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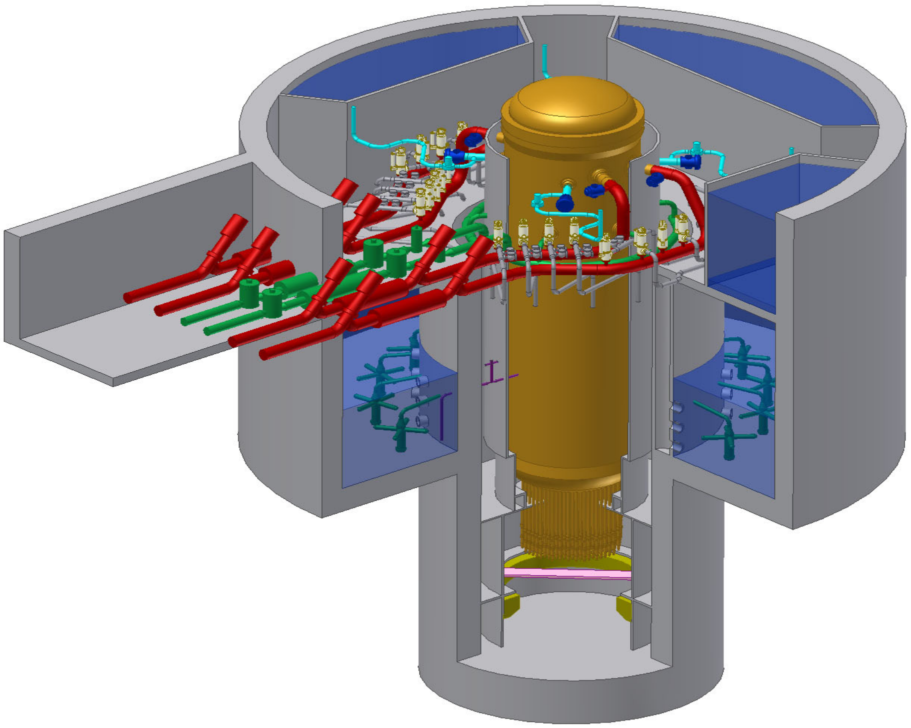

ESBWR Design Control Document

Tier 2

Chapter 12

Radiation Protection

(Conditional Release - pending closure of design verifications)

Contents

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Abbreviations And Acronyms

Term	Definition
10 CFR	Title 10, Code of Federal Regulations
A/D	Analog-to-Digital
AASHTO	American Association of Highway and Transportation Officials
AB	Auxiliary Boiler
ABS	Auxiliary Boiler System
ABWR	Advanced Boiling Water Reactor
ac / AC	Alternating Current
AC	Air Conditioning
ACF	Automatic Control Function
ACI	American Concrete Institute
ACS	Atmospheric Control System
AD	Administration Building
ADS	Automatic Depressurization System
AEC	Atomic Energy Commission
AFIP	Automated Fixed In-Core Probe
AGMA	American Gear Manufacturer's Association
AHS	Auxiliary Heat Sink
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
AL	Analytical Limit
ALARA	As Low As Reasonably Achievable
ALWR	Advanced Light Water Reactor
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
AOV	Air Operated Valve
API	American Petroleum Institute
APRM	Average Power Range Monitor
APR	Automatic Power Regulator
APRS	Automatic Power Regulator System
ARI	Alternate Rod Insertion
ARMS	Area Radiation Monitoring System
ASA	American Standards Association
ASD	Adjustable Speed Drive
ASHRAE	American Society of Heating, Refrigerating, and Air Conditioning Engineers
ASME	American Society of Mechanical Engineers
AST	Alternate Source Term
ASTM	American Society of Testing Methods
AT	Unit Auxiliary Transformer

Term	Definition
ATLM	Automated Thermal Limit Monitor
ATWS	Anticipated Transients Without Scram
AV	Allowable Value
AWS	American Welding Society
AWWA	American Water Works Association
B&PV	Boiler and Pressure Vessel
BAF	Bottom of Active Fuel
BHP	Brake Horse Power
BOP	Balance of Plant
BPU	Bypass Unit
BPV	Bypass Valve
BPWS	Banked Position Withdrawal Sequence
BRE	Battery Room Exhaust
BRL	Background Radiation Level
BTP	NRC Branch Technical Position
BTU	British Thermal Unit
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CAV	Cumulative absolute velocity
C&FS	Condensate and Feedwater System
C&I	Control and Instrumentation
C/C	Cooling and Cleanup
CB	Control Building
CBGAHVS	Control Building General Area
CBHVAC	Control Building HVAC
CBHVS	Control Building Heating, Ventilation and Air Conditioning System
CCI	Core-Concrete Interaction
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CIRC	Circulating Water System
CIS	Containment Inerting System
CIV	Combined Intermediate Valve
CLAVS	Clean Area Ventilation Subsystem of Reactor Building HVAC
CM	Cold Machine Shop
CMS	Containment Monitoring System
CMU	Control Room Multiplexing Unit
COL	Combined Operating License
COLR	Core Operating Limits Report
CONAVS	Controlled Area Ventilation Subsystem of Reactor Building HVAC
CPR	Critical Power Ratio

Term	Definition
CPS	Condensate Purification System
CPU	Central Processing Unit
CR	Control Rod
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CRDH	Control Rod Drive Housing
CRDHS	Control Rod Drive Hydraulic System
CRGT	Control Rod Guide Tube
CRHA	Control Room Habitability Area
CRHAHVS	Control Room Habitability Area HVAC Sub-system
CRT	Cathode Ray Tube
CS&TS	Condensate Storage and Transfer System
CSDM	Cold Shutdown Margin
CS / CST	Condensate Storage Tank
CT	Main Cooling Tower
CTVCF	Constant Voltage Constant Frequency
CUF	Cumulative usage factor
CWS	Chilled Water System
D-RAP	Design Reliability Assurance Program
DAC	Design Acceptance Criteria
DAW	Dry Active Waste
DBA	Design Basis Accident
DBE	Design Basis Event
dc / DC	Direct Current
DCS	Drywell Cooling System
DCIS	Distributed Control and Information System
DEPSS	Drywell Equipment and Pipe Support Structure
DF	Decontamination Factor
D/F	Diaphragm Floor
DG	Diesel-Generator
DHR	Decay Heat Removal
DM&C	Digital Measurement and Control
DOF	Degree of freedom
DOI	Dedicated Operators Interface
DOT	Department of Transportation
dPT	Differential Pressure Transmitter
DPS	Diverse Protection System
DPV	Depressurization Valve
DR&T	Design Review and Testing
DS	Independent Spent Fuel Storage Installation

Term	Definition
DTM	Digital Trip Module
DW	Drywell
EB	Electrical Building
EBAS	Emergency Breathing Air System
EBHV	Electrical Building HVAC
ECCS	Emergency Core Cooling System
E-DCIS	Essential DCIS (Distributed Control and Information System)
EDO	Environmental Qualification Document
EFDS	Equipment and Floor Drainage System
EFPY	Effective full power years
EFU	Emergency Filter Unit
EHC	Electrohydraulic Control (Pressure Regulator)
ENS	Emergency Notification System
EOC	Emergency Operations Center
EOC	End of Cycle
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedures
EPDS	Electric Power Distribution System
EPG	Emergency Procedure Guidelines
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
ERICP	Emergency Rod Insertion Control Panel
ERIP	Emergency Rod Insertion Panel
ESF	Engineered Safety Feature
ETS	Emergency Trip System
FAC	Flow-Accelerated Corrosion
FAPCS	Fuel and Auxiliary Pools Cooling System
FATT	Fracture Appearance Transition Temperature
FB	Fuel Building
FBHV	Fuel Building HVAC
FCI	Fuel-Coolant Interaction
FCM	File Control Module
FCS	Flammability Control System
FCU	Fan Cooling Unit
FDDI	Fiber Distributed Data Interface
FFT	Fast Fourier Transform
FFWTR	Final Feedwater Temperature Reduction
FHA	Fire Hazards Analysis
FIV	Flow-Induced Vibration
FMCRD	Fine Motion Control Rod Drive

Term	Definition
FMEA	Failure Modes and Effects Analysis
FPS	Fire Protection System
FO	Diesel Fuel Oil Storage Tank
FOAKE	First-of-a-Kind Engineering
FPE	Fire Pump Enclosure
FTDC	Fault-Tolerant Digital Controller
FTS	Fuel Transfer System
FW	Feedwater
FWCS	Feedwater Control System
FWS	Fire Water Storage Tank
GCS	Generator Cooling System
GDC	General Design Criteria
GDCS	Gravity-Driven Cooling System
GE	General Electric Company
GE-NE	GE Nuclear Energy
GEN	Main Generator System
GETAB	General Electric Thermal Analysis Basis
GL	Generic Letter
GM	Geiger-Mueller Counter
GM-B	Beta-Sensitive GM Detector
GSIC	Gamma-Sensitive Ion Chamber
GSOS	Generator Sealing Oil System
GWSR	Ganged Withdrawal Sequence Restriction
HAZ	Heat-Affected Zone
HCU	Hydraulic Control Unit
HCW	High Conductivity Waste
HDVS	Heater Drain and Vent System
HEI	Heat Exchange Institute
HELB	High Energy Line Break
HEP	Human error probability
HEPA	High Efficiency Particulate Air/Absolute
HFE	Human Factors Engineering
HFF	Hollow Fiber Filter
HGCS	Hydrogen Gas Cooling System
HIC	High Integrity Container
HID	High Intensity Discharge
HIS	Hydraulic Institute Standards
HM	Hot Machine Shop & Storage
HP	High Pressure
HPNSS	High Pressure Nitrogen Supply System

Term	Definition
HPT	High-pressure turbine
HRA	Human Reliability Assessment
HSI	Human-System Interface
HSSS	Hardware/Software System Specification
HVAC	Heating, Ventilation and Air Conditioning
HVS	High Velocity Separator
HWC	Hydrogen Water Chemistry
HWCS	Hydrogen Water Chemistry System
HWS	Hot Water System
HX	Heat Exchanger
I&C	Instrumentation and Control
I/O	Input/Output
IAS	Instrument Air System
IASCC	Irradiation Assisted Stress Corrosion Cracking
IBC	International Building Code
IC	Ion Chamber
IC	Isolation Condenser
ICD	Interface Control Diagram
ICS	Isolation Condenser System
IE	Inspection and Enforcement
IEB	Inspection and Enforcement Bulletin
IED	Instrument and Electrical Diagram
IEEE	Institute of Electrical and Electronic Engineers
IFTS	Inclined Fuel Transfer System
IGSCC	Intergranular Stress Corrosion Cracking
IIS	Iron Injection System
ILRT	Integrated Leak Rate Test
IOP	Integrated Operating Procedure
IMC	Induction Motor Controller
IMCC	Induction Motor Controller Cabinet
IRM	Intermediate Range Monitor
ISA	Instrument Society of America
ISI	In-Service Inspection
ISLT	In-Service Leak Test
ISM	Independent Support Motion
ISMA	Independent Support Motion Response Spectrum Analysis
ISO	International Standards Organization
ITA	Inspections, Tests or Analyses
ITAAC	Inspections, Tests, Analyses and Acceptance Criteria
ITA	Initial Test Program

Term	Definition
LAPP	Loss of Alternate Preferred Power
LCO	Limiting Conditions for Operation
LCW	Low Conductivity Waste
LD	Logic Diagram
LDA	Lay down Area
LD&IS	Leak Detection and Isolation System
LERF	Large early release frequency
LFCV	Low Flow Control Valve
LHGR	Linear Heat Generation Rate
LLRT	Local Leak Rate Test
LMU	Local Multiplexer Unit
LO	Dirty/Clean Lube Oil Storage Tank
LOCA	Loss-of-Coolant-Accident
LOFW	Loss-of-feedwater
LOOP	Loss of Offsite Power
LOPP	Loss of Preferred Power
LP	Low Pressure
LPCI	Low Pressure Coolant Injection
LPCRD	Locking Piston Control Rod Drive
LPMS	Loose Parts Monitoring System
LPRM	Local Power Range Monitor
LPSP	Low Power Setpoint
LWMS	Liquid Waste Management System
MAAP	Modular Accident Analysis Program
MAPLHGR	Maximum Average Planar Linear Head Generation Rate
MAPRAT	Maximum Average Planar Ratio
MBB	Motor Built-In Brake
MCC	Motor Control Center
MCES	Main Condenser Evacuation System
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room
MCRP	Main Control Room Panel
MELB	Moderate Energy Line Break
MLHGR	Maximum Linear Heat Generation Rate
MMI	Man-Machine Interface
MMIS	Man-Machine Interface Systems
MOV	Motor-Operated Valve
MPC	Maximum Permissible Concentration
MPL	Master Parts List
MS	Main Steam

Term	Definition
MSIV	Main Steam Isolation Valve
MSL	Main Steamline
MSLB	Main Steamline Break
MSLBA	Main Steamline Break Accident
MSR	Moisture Separator Reheater
MSV	Mean Square Voltage
MT	Main Transformer
MTTR	Mean Time To Repair
MWS	Makeup Water System
NBR	Nuclear Boiler Rated
NBS	Nuclear Boiler System
NCIG	Nuclear Construction Issues Group
NDE	Nondestructive Examination
NE-DCIS	Non-Essential Distributed Control and Information System
NDRC	National Defense Research Committee
NDT	Nil Ductility Temperature
NFPA	National Fire Protection Association
NIST	National Institute of Standard Technology
NICWS	Nuclear Island Chilled Water Subsystem
NMS	Neutron Monitoring System
NOV	Nitrogen Operated Valve
NPHS	Normal Power Heat Sink
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NRHX	Non-Regenerative Heat Exchanger
NS	Non-seismic (non-seismic Category I)
NSSS	Nuclear Steam Supply System
NT	Nitrogen Storage Tank
NTSP	Nominal Trip Setpoint
O&M	Operation and Maintenance
O-RAP	Operational Reliability Assurance Program
OBCV	Overboard Control Valve
OBE	Operating Basis Earthquake
OGS	Offgas System
OHLHS	Overhead Heavy Load Handling System
OIS	Oxygen Injection System
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLU	Output Logic Unit
OOS	Out-of-service
ORNL	Oak Ridge National Laboratory

Term	Definition
OSC	Operational Support Center
OSHA	Occupational Safety and Health Administration
OSI	Open Systems Interconnect
P&ID	Piping and Instrumentation Diagram
PA/PL	Page/Party-Line
PABX	Private Automatic Branch (Telephone) Exchange
PAM	Post Accident Monitoring
PAR	Passive Autocatalytic Recombiner
PAS	Plant Automation System
PASS	Post Accident Sampling Subsystem of Containment Monitoring System
PCC	Passive Containment Cooling
PCCS	Passive Containment Cooling System
PCT	Peak cladding temperature
PCV	Primary Containment Vessel
PFD	Process Flow Diagram
PGA	Peak Ground Acceleration
PGCS	Power Generation and Control Subsystem of Plant Automation System
PH	Pump House
PL	Parking Lot
PM	Preventive Maintenance
PMCS	Performance Monitoring and Control Subsystem of NE-DCIS
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
PQCL	Product Quality Check List
PRA	Probabilistic Risk Assessment
PRMS	Process Radiation Monitoring System
PRNM	Power Range Neutron Monitoring
PS	Plant Stack
PSD	Power Spectra Density
PSS	Process Sampling System
PSWS	Plant Service Water System
PT	Pressure Transmitter
PWR	Pressurized Water Reactor
QA	Quality Assurance
RACS	Rod Action Control Subsystem
RAM	Reliability, Availability and Maintainability
RAPI	Rod Action and Position Information
RAT	Reserve Auxiliary Transformer
RB	Reactor Building
RBC	Rod Brake Controller

Term	**Definition**
RBCC	Rod Brake Controller Cabinet
RBCWS	Reactor Building Chilled Water Subsystem
RBHV	Reactor Building HVAC
RBS	Rod Block Setpoint
RBV	Reactor Building Vibration
RC&IS	Rod Control and Information System
RCC	Remote Communication Cabinet
RCCV	Reinforced Concrete Containment Vessel
RCCWS	Reactor Component Cooling Water System
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RDA	Rod Drop Accident
RDC	Resolver-to-Digital Converter
REPAVS	Refueling and Pool Area Ventilation Subsystem of Fuel Building HVAC
RFP	Reactor Feed Pump
RG	Regulatory Guide
RHR	Residual heat removal (function)
RHX	Regenerative Heat Exchanger
RMS	Root Mean Square –or- Radiation Monitoring Subsystem
RMU	Remote Multiplexer Unit
RO	Reverse Osmosis
ROM	Read-only Memory
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RRPS	Reference Rod Pull Sequence
RSM	Rod Server Module
RSPC	Rod Server Processing Channel
RSS	Remote Shutdown System
RSSM	Reed Switch Sensor Module
RSW	Reactor Shield Wall
RTIF	Reactor Trip and Isolation Function(s)
RT_{NDT}	Reference Temperature of Nil-Ductility Transition
RTP	Reactor Thermal Power
RW	Radwaste Building
RWBCR	Radwaste Building Control Room
RWBGA	Radwaste Building General Area
RWBHVAC	Radwaste Building HVAC
RWCU/SDC	Reactor Water Cleanup/Shutdown Cooling
RWE	Rod Withdrawal Error
RWM	Rod Worth Minimizer

Term	Definition
SA	Severe Accident
SAR	Safety Analysis Report
SB	Service Building
S/C	Digital Gamma-Sensitive GM Detector
SC	Suppression Chamber
S/D	Scintillation Detector
S/DRSRO	Single/Dual Rod Sequence Restriction Override
S/N	Signal-to-Noise
S/P	Suppression Pool
SAS	Service Air System
SB&PC	Steam Bypass and Pressure Control System
SBO	Station Blackout
SBWR	Simplified Boiling Water Reactor
SCEW	System Component Evaluation Work
SCRRI	Selected Control Rod Run-in
SDC	Shutdown Cooling
SDM	Shutdown Margin
SDS	System Design Specification
SEOA	Sealed Emergency Operating Area
SER	Safety Evaluation Report
SF	Service Water Building
SFP	Spent fuel pool
SIL	Service Information Letter
SIT	Structural Integrity Test
SIU	Signal Interface Unit
SJAE	Steam Jet Air Ejector
SLC	Standby Liquid Control
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SMU	SSLC Multiplexing Unit
SOV	Solenoid Operated Valve
SP	Setpoint
SPC	Suppression Pool Cooling
SPDS	Safety Parameter Display System
SPTMS	Suppression Pool Temperature Monitoring Subsystem of Containment Monitoring System
SR	Surveillance Requirement
SRM	Source Range Monitor
SRNM	Startup Range Neutron Monitor
SRO	Senior Reactor Operator
SRP	Standard Review Plan

Term	**Definition**
SRS	Software Requirements Specification
SRSRO	Single Rod Sequence Restriction Override
SRSS	Sum of the squares
SRV	Safety Relief Valve
SRVDL	Safety relief valve discharge line
SSAR	Standard Safety Analysis Report
SSC(s)	Structure, System and Component(s)
SSE	Safe Shutdown Earthquake
SSLC	Safety System Logic and Control
SSPC	Steel Structures Painting Council
ST	Spare Transformer
STI	Startup Test Instruction
STP	Sewage Treatment Plant
STRAP	Scram Time Recording and Analysis Panel
STRP	Scram Time Recording Panel
SV	Safety Valve
SWH	Static water head
SWMS	Solid Waste Management System
SY	Switch Yard
TAF	Top of Active Fuel
TASS	Turbine Auxiliary Steam System
TB	Turbine Building
TBCE	Turbine Building Compartment Exhaust
TEAS	Turbine Building Air Supply
TBE	Turbine Building Exhaust
TBLOE	Turbine Building Lube Oil Area Exhaust
TBS	Turbine Bypass System
TBHV	Turbine Building HVAC
TBV	Turbine Bypass Valve
TC	Training Center
TCCWS	Turbine Component Cooling Water System
TCS	Turbine Control System
TCV	Turbine Control Valve
TDH	Total Developed Head
TEMA	Tubular Exchanger Manufacturers' Association
TFSP	Turbine first stage pressure
TG	Turbine Generator
TGSS	Turbine Gland Seal System
THA	Time-history accelerograph
TLOS	Turbine Lubricating Oil System

Term	Definition
TLU	Trip Logic Unit
TMI	Three Mile Island
TMSS	Turbine Main Steam System
TRM	Technical Requirements Manual
TS	Technical Specification(s)
TSC	Technical Support Center
TSI	Turbine Supervisory Instrument
TSV	Turbine Stop Valve
TTWFATBV	Turbine trip with failure of all bypass valves
UBC	Uniform Building Code
UHS	Ultimate heat sink
UL	Underwriter's Laboratories Inc.
UPS	Uninterruptible Power Supply
USE	Upper Shelf Energy
USM	Uniform Support Motion
USMA	Uniform support motion response spectrum analysis
USNRC	United States Nuclear Regulatory Commission
USS	United States Standard
UV	Ultraviolet
V&V	Verification and Validation
Vac / VAC	Volts Alternating Current
Vdc / VDC	Volts Direct Current
VDU	Video Display Unit
VW	Vent Wall
VWO	Valves Wide Open
WD	Wash Down Bays
WH	Warehouse
WS	Water Storage
WT	Water Treatment
WW	Wetwell
XMFR	Transformer
ZPA	Zero period acceleration

12. RADIATION PROTECTION

12.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE ALARA

12.1.1 Policy Considerations

Administrative programs and procedures, in conjunction with facility design, ensure that the occupational radiation exposure to personnel is kept as low as reasonably achievable (ALARA).

12.1.1.1 Design and Construction Policies

The ALARA philosophy was applied during the initial design of the plant and implemented via internal design reviews. The design was reviewed in detail for ALARA considerations and was reviewed, updated and modified as necessary during the design phase as experience was gained from operating plants. Engineers reviewed the plant design and integrated the layout, shielding, ventilation and monitoring instrument designs with traffic control, security, access control, and health physics aspects to ensure the overall design is conducive to maintaining exposures ALARA.

All pipe routing containing radioactive fluids was reviewed as part of the engineering design effort. This ensured that lines expected to contain significant radiation sources are adequately shielded and properly routed to minimize exposure to personnel.

Operating plant results were continuously integrated during the design phase of the ESBWR Standard Plant.

12.1.1.2 Operation Policies

Out of ESBWR Standard Plant scope. The Combined Operating Licensing applicant will address the activities conducted by management personnel who have plant operational responsibility for radiation protection.

12.1.1.3 Compliance with 10 CFR 20 and Regulatory Guides 8.8, 8.10 and 1.8

Compliance of the ESBWR design with Title 10 of the Code of Federal Regulations, Part 20 (10 CFR 20), is ensured by the compliance of the design and operation of the facility within the guidelines of Regulatory Guides 8.8, 8.10, and 1.8.

12.1.1.3.1 Compliance with Regulatory Guide 8.8

The policy considerations regarding plant operations contained in Regulatory Guide 8.8 are out of ESBWR Standard Plant Scope. See Subsection 12.1.4.4 for COL license information.

The design of ESBWR plant meets the guidelines of Regulatory Guide 8.8, Sections C.2 and C.4, which address facility, equipment and instrumentation design features. Features of the plant that are examples of compliance with Regulatory Guide 8.8 are delineated in Section 12.3.

12.1.1.3.2 Compliance with Regulatory Guide 8.10

Out of ESBWR Standard Plant scope. See Subsection 12.1.4.1 for COL license information.

12.1.1.3.3 Compliance with Regulatory Guide 1.8

Out of ESBWR Standard Plant scope. See Subsection 12.1.4.2 for COL license information.

12.1.2 Design Considerations

This subsection discusses the methods and features by which the policy considerations of Subsection 12.1.1 are applied. Provisions and designs for maintaining personnel exposures ALARA are presented in detail in Subsections 12.3.1 and 12.3.2.

12.1.2.1 General Design Consideration for ALARA Exposures

General design considerations and methods employed to maintain in-plant radiation exposures ALARA, consistent with the recommendations of Regulatory Guide 8.8, have two objectives:

- Minimizing the necessity for and amount of personnel time spent in radiation areas, and
- Minimizing radiation levels in routinely occupied plant areas in the vicinity of plant equipment expected to require personnel attention.

Both equipment and facility designs are considered in maintaining exposures ALARA during plant operations. Events considered include normal operation maintenance and repairs, refueling operations and fuel storage, in-service inspection and calibrations, radioactive waste handling and disposal, etc.

The features of the plant design that ensure the plant can be operated and maintained with ALARA exposures also serve to assist in achieving ALARA exposures during the decommissioning process.

Examples of features that assist in maintaining low occupational exposures during decommissioning include the following:

- Provisions for draining, flushing, and decontaminating equipment and piping.
- Design of equipment to minimize the buildup of radioactive material and to facilitate flushing of crud traps.
- Shielding which provides protection during maintenance or repairs and during decommissioning operations.
- Provision of means and adequate space for utilization of movable shielding.
- Separation of more highly radioactive equipment from less radioactive equipment and provision of separate shielded compartments for adjacent items of radioactive equipment.
- Provision for access hatches for the installation or removal of plant components.
- Provision of design features such as the Reactor Water Cleanup/Shutdown Cooling System and the condensate demineralizer to minimize crud buildup.

12.1.2.2 Equipment Design Considerations for ALARA Exposures

12.1.2.2.1 General Design Criteria

No specific instructions have been given to component designers and engineers regarding ALARA design as provided by specific Acceptance Criterion II.2 of SRP Section 12.1. However, the engineering design procedures require that the component design engineer consider the applicable Regulatory Guides (including Regulatory Guide 8.8) as a part of the design criteria. In this way, the radiation problems of a component or system are considered. A summary survey of the components designs was made to determine the factors considered. The following paragraphs cite some examples of design considerations made to implement ALARA.

12.1.2.2.2 Equipment Design Considerations to Limit Time Spent in Radiation Areas

Equipment is designed to be operated and have its instrumentation and controls in accessible areas both during normal and abnormal operating conditions. Equipment such as the Reactor Water Cleanup/Shutdown Cooling System (RWCU/SDC) and the Fuel and Auxiliary Pool Cooling System (FAPCS) are remotely operated, including the backwashing and precoat operations.

Equipment is designed to facilitate maintenance. Equipment such as the Isolation Condenser heat exchanger is designed with an excess of tubes in order to permit plugging of some tubes. The heat exchanger has drains to allow draining of the shell-side water. Some of the valves have stem packing of the cartridge type that can be easily replaced. Refueling tools are designed for drainage and with smooth surfaces in order to reduce contamination. Vessel and piping insulation is of an easily removable type.

The materials selected for use in the system have been chosen to fulfill environmental requirements. Valves, for example, use grafoil stem packing to reduce leakage and maintenance.

Past experience has been factored into current designs. The steam relief valves have been redesigned as a result of inservice testing.

12.1.2.2.3 Equipment Design Considerations to Limit Component Radiation Levels

Equipment and piping were designed to reduce the accumulation of radioactive materials in the equipment. The piping, where possible, was constructed of seamless pipe as a means to reduce radiation accumulation on the seam. The filter demineralizers in the RWCU/SDC and FAPCS are backwashed and flushed prior to maintenance.

Equipment designs include provisions for limiting leaks or controlling the fluid that does leak. This includes piping the released fluid to the sumps and using drip pans with drains piped to the floor drains.

The materials selected for use in the primary coolant system consist mainly of austenitic stainless steel, carbon steel and low alloy steel components.

The system design includes a RWCU/SDC on the reactor coolant. This system is designed to limit the radioactive isotopes in the coolant.

12.1.2.3 Facility Layout General Design Considerations for Maintaining Radiation Exposures ALARA

12.1.2.3.1 Minimizing Personnel Time Spent in Radiation Areas

Facility general design considerations to minimize the amount of personnel time spent in radiation areas include the following:

- Locating equipment, instruments, and sampling stations, that require routine maintenance, calibration, operation, or inspection, for ease of access and minimum required occupancy time in radiation areas
- Laying out plant areas to allow remote or mechanical operation, service, monitoring, or inspection of highly radioactive equipment
- Providing, where practicable, for transportation of equipment or components requiring service to a lower radiation area

12.1.2.3.2 Minimizing Radiation Levels in Plant Access Areas and Vicinity of Equipment

Facility general design considerations directed toward minimizing radiation levels in plant access areas and in the vicinity of equipment requiring personnel attention include the following:

- Separating radiation sources and occupied areas where practicable (e.g., pipes or ducts containing potentially high radioactive fluids not passing through occupied areas).
- Providing adequate shielding between radiation sources and access and service areas. Of special note, the reactor pressure vessel shield wall in the upper drywell extends to within half a meter of the upper drywell ceiling, thus permitting continued operation in the upper drywell during refueling and providing shielding in the case of a refueling accident.
- Locating equipment, instruments, and sampling sites in the lowest practicable radiation zone.
- Providing central control panels to permit remote operation of all essential instrumentation and controls from the lowest radiation zone practicable.
- Where practicable for package units, separating highly radioactive equipment from less radioactive equipment, instruments, and controls.
- Providing means and adequate space for utilizing moveable shielding for sources within the service area when required.
- Providing means to control contamination and to facilitate decontamination of potentially contaminated areas where practicable.
- Providing means for decontamination of service areas.
- Providing space for pumps and valves outside of highly radioactive areas.
- Providing remotely-operated centrifugal discharge and/or back-flushable filter systems for highly radioactive radwaste and cleanup systems.
- Providing labyrinth entrances to radioactive pump, equipment, and valve rooms.
- Providing adequate space in labyrinth entrances for easy access.

- Maintaining ventilation airflow patterns from areas of lower radioactivity to areas of higher radioactivity.

12.1.3 Operational Considerations

Out of ESBWR Standard Plant scope. See Subsection 12.1.4.3 for COL license information.

12.1.4 COL Information

12.1.4.1 Regulatory Guide 8.10

The COL applicant shall demonstrate Compliance with Regulatory Guide 8.10 (Subsection 12.1.1.3.2).

12.1.4.2 Regulatory Guide 1.8

The COL applicant shall demonstrate Compliance with Regulatory Guide 1.8 (Subsection 12.1.1.3.3).

12.1.4.3 Occupational Radiation Exposures

COL applicants will provide, to the level of detail provided in Regulatory Guide 1.70, the criteria and/or conditions under which various operating procedures and techniques shall be provided to ensure that occupational radiation exposures ALARA are implemented (Subsection 12.1.3).

12.1.4.4 Regulatory Guide 8.8

The COL applicant will demonstrate compliance with Regulatory Guide 8.8 (Subsection 12.1.1.3.1).

12.1.5 References

None.

12.2 PLANT SOURCES

12.2.1 Contained Sources

12.2.1.1 Primary Containment Source Terms

This section provides a summation of the significant radioactive source terms found in the ESBWR containment. These source terms consist of those elements which are found to contain significant quantities of radioactive materials but do not include sources due to incidental contamination such as sources in valves due to deposition of corrosion or fission product species on the surfaces of the components. As such, the ESBWR unlike prior BWRs has only one significant source of radiation in the containment post operation, the reactor core. In addition, the Fine Motion Control Rod Drive (FMCRD) System provides the only other notable source of radiation in the containment. The ESBWR does not contain any recirculation pumps (external or internal), Traversing Incore Probe system, or heat exchangers that, as a function of normal usage, may become contaminated. Subsection 12.2.1.1 discusses the design of and sources found in the reactor core, while Subsection 12.2.1.2 discusses the Reactor Building source terms.

12.2.1.1.1 Reactor Vessel Core Sources

The information in this section defines a reactor vessel model and pertinent data necessary to calculate neutron and gamma fluxes inside and outside of the reactor core during normal operation. Calculation of excore particle fluxes from the reactor core during operation requires a detailed analysis of neutral particle transport, and, hence, requires the use of either a deterministic solution to the Boltzmann equation or the use of probabilistic modeling techniques. The primary source for both the neutron and gamma fluxes outside of the core is the fission process. Gammas are also created by the decay of fission products, and secondary gammas resulting from neutron absorption and scattering in structural materials both inside and outside of the core. Nuclide cross-section libraries contain gamma production data for all of these sources; therefore, it is necessary only to define the neutron fission source in the core, and then to perform a coupled neutron-gamma transport calculation. The data in this section is intended to supply adequate information to generate a neutron fission source and define geometric regions sufficient to perform a fixed source calculation using either of the methods.

Also contained in this section are post-operation gamma sources in the containment. After shutdown, the neutron fluxes are negligible and nitrogen-16 quickly decays to zero. Therefore, the most significant source is the gammas resulting from fission product decay in the reactor core.

Physical Data

Table 12.2-1 presents the physical data required to form the model in Figure 12.2-1. This model was selected to provide sufficient regions to adequately portray the reactor. The incore region was divided into 25 axial nodes, with one radial node per fuel bundle. A unique neutron fission source was determined for each of these nodes using the nodal cycle average power and exposure data. Water densities were determined at each of the 25 planes for peripheral bundles and in-core bundles. Table 12.2-1 provides nominal dimensions and material volume fractions for each boundary and region in the reactor model with core average data presented for the core. To describe the reactor core, Table 12.2-1 provides thermal power, power density, core dimensions, core average material volume fractions, and cycle average reactor power distributions and exposures. The reactor power distributions are given for both radial and axial distributions and represent the cycle averages for an equilibrium cycle.

Core Boundary and Vessel Neutron Fluxes

Table 12.2-2 presents multigroup neutron fluxes at the representative location of the core boundary and at the vessel. The multigroup neutron fluxes and the fast neutron flux (E>1 MeV) at the peak elevation of the core boundary, vessel inside surface, and ¼ thickness of the vessel are presented in Table 12.2-2, Part A. The uncertainty of the fast neutron flux at the vessel is estimated to be within ±19%. Normalized axial variations for the fast flux at the vessel inside surface are shown in Part B of Table 12.2-2.

Gamma Ray Source Energy Spectra

Table 12.2-3 presents the average gamma ray source energy spectra in both core and non-core regions. In Table 12.2-3, Part A, the energy spectrum in the core, bypass water, shroud, downcomer, and RPV is presented. This represents the average gamma ray energy released by energy group per unit volume of the region. The energy spectra in MeV per sec per cm^3 can be used with the power distributions to obtain the source in any part of the core.

The gamma ray energy spectrum includes the fission gamma rays, the fission product gamma rays, and the gamma rays resulting from inelastic neutron scattering and neutron capture. The total gamma ray energy released in the core is estimated to be accurate to within ±20%.

Post-Operation Gamma Sources

Table 12.2-3 Part B gives a gamma ray energy spectrum in MeV/sec per MW thermal in spent fuel as a function of time after operation. The data were prepared from the irradiation and decay calculation of a representative ESBWR fuel bundle to an average exposure of 35 GWd/MTU. To obtain shutdown sources in the core, the gamma ray energy spectra are combined with the core thermal power and power distributions. Shutdown sources in a single fuel element can be obtained by using the gamma ray energy spectra and the thermal power of the element during operation.

Gamma Ray and Neutron Fluxes Outside the Vessel

Table 12.2-4 presents the maximum axial neutron and gamma ray fluxes outside the vessel. The maximum axial flux occurs typically near the core midplane elevation where the maximum power density is located for the peripheral bundle. This elevation can be located using the data from Table 12.2-1. The fluxes at this elevation represent the fluxes at the peak azimuth angle. The gamma ray calculations include gamma ray sources from all regions inside the vessel and the vessel itself.

12.2.1.1.2 Other Radioactive Sources

Radioactive Sources in the Control Rod Drive System

The control rod drive (CRD) source term data are provided in Table 12.2-5. The system is described in Subsection 3.9.4.

Reactor Startup Source

The reactor startup source is shipped to the site in a special cask designed with shielding. The source is transferred under water while in the cask and loaded into beryllium containers. This is then loaded into the reactor while remaining under water. The source remains within the reactor for its lifetime. Thus, no unique shielding requirements are required after reactor operation.

12.2.1.2 Reactor Building Source Terms

This section provides a summation of the significant radioactive source terms found in the ESBWR reactor building. These source terms consist of those elements which are found to contain significant quantities of radioactive materials but do not include sources due to incidental contamination such as sources in valves and pipes due to deposition of corrosion or fission products species on the surfaces of the components.

The reactor building (RB) is divided into three specific zones:

- Containment
- Contaminated areas,
- Clean areas.

Radioactive Sources in the Reactor Water Cleanup/Shutdown Cooling System.

A description of the Reactor Water Cleanup/Shutdown Cooling System (RWCU/SDC) is given in Subsection 5.4.8. Radioactive sources contained in this system are the result of contamination of components by transit of reactor water through this system and accumulation of radioisotopes removed from the water. Components for this system include regenerative and non-regenerative heat exchangers, pumps, valves, and demineralizers. The accumulated sources in this system are given in Tables 12.2-6 and 12.2-7. The sources present in the demineralizers are present in all modes of operation. Therefore, backwashing capability is provided to remove residual activity with clean water plus chemical decontamination for effective radwaste handling.

12.2.1.2.1 Other Sources

Radioactive Sources in the Fuel and Auxiliary Pools Cooling System (FAPCS)

A description of the FAPCS is given in Subsection 9.1.3. The FAPCS is designed to service the fuel pools, suppression pool, GDCS pool, and isolation condenser/PCCS pools on a rotating basis. The accumulated activity in this system is the result of the accumulation of residual activity in each of the above pools. The filters are backwashed into a backwash receiving tank, which is then routed to the Radwaste Building systems. The sources for the FAPCS are given in Tables 12.2-8 and 12.2-9. Clean water connections are provided for this system to flush lines prior to switching between pools as necessary to prevent ancillary contamination between pools.

Radioactive Sources in the Spent Fuel Pool

The radiation sources in the spent fuel pool are given in Table 12.2-3 Part B in terms of MeV/sec-MWt. Water concentration is assumed as 1% of normal reactor water concentration (Section 11.1).

Radioactive Sources in the HVAC System

The HVAC System is described in Section 9.4 and employs a bypass HEPA filter train for use in the event of airborne contamination of the RB or controlled purge of the RB containment. The HEPA train is capable of removing all large particulate releases and up to 70% of small particulate releases. As such, no significant radioactive contamination of the HEPA train is expected.

Radioactive Sources in the Main Steam and Feedwater Lines

All radioactive material in the main steam system result from radioactive sources carried over from the reactor core during plant operations. In most components carrying live steam, N^{16} is the dominant source of radioactivity (Section 11.1). Otherwise, under conditions where sufficient decay time has removed the N^{16} source, noble radiogases become the dominant source term (Section 11.1). Flow in the feedwater lines is dominated by corrosion and fission products and is the result of the residual activity of reactor steam after treatment in the condenser filter-demineralizer system.

Post-Accident Radioactive Sources

The ESBWR design limits potential radiation exposure from accidents both to plant personnel and to the public by the use of passive safety features, containment and treatment of potential accident sources. The following describes those features of the ESBWR germane to post-accident radiation sources in the RB containment and the RB.

The RB containment is an inert steel-lined pressure boundary capable of containing all accident sources with minimal leakage to the environment or other plant areas. The containment is provided with redundant passive cooling systems (Subsections 5.4.6 and 6.2.2) to insure within a reasonable probability that this primary boundary does not exceed design criteria. Drywell spray provides additional capability to control pressure. Therefore, for all but the most improbable accident scenarios requiring massive failures of all major systems including passive systems, radioactive sources from the pressure vessel are adequately contained in the RB containment.

Surrounding the containment on all sides, the ESBWR employs a RB that provides a secondary holdup volume (Subsection 6.2.3) to trap containment penetration and valve leakage except direct bypass leakage via such lines as the main steam lines and feedwater lines. All major connections from the containment, except the isolation condenser steam lines and condensate lines and the main steam lines and feedwater lines requiring isolation valves, terminate with the second isolation valve in the RB. The RWCU/SDC is the only high energy line in the containment and RB that could produce potential releases in the containment or RB. High energy line rupture releases in the containment are isolated by the HVAC system for holdup and treatment, except potential high energy breaks, which are then routed to the turbine building for release via the plant stack. High energy line rupture releases in the RB are routed to the refueling floor where a rupture disk relieves the overpressure. See Section 15.4 for discussions of line break releases.

Estimates on sources and location for limiting design basis events are found in Section 15.4.

12.2.1.3 Turbine Building Source Terms

This section provides a summation of the significant radioactive source terms found in the ESBWR turbine building. These source terms consist of those elements which are found to contain significant quantities of radioactive materials but do not include sources due to incidental contamination such as sources in valves due to deposition of corrosion or fission products species on the surfaces of the components.

Normal Operating Sources

N^{16} in the steam flow from the pressure vessel. is the primary turbine building source of radioactivity. The N^{16} source results in significant gamma shine from the main steam lines and steam bearing components on the order of 0.2-0.5 Gy/hr (20-50 rad/hr) contact. Other major sources of radiation in the turbine building are the Offgas System (Section 11.3) and the Condenser and Feedwater System. The Offgas System consists of the steam jet air ejector, recombiner, offgas condenser, and offgas charcoal tanks. Table 12.2-10 provides the sources for the Offgas System. The sources for the turbine condenser and feedwater filter/demineralizer system are given in Tables 12.2-11 and 12.2-12.

Post-Accident Radioactive Sources

The turbine building contains no major sources of releasable radioactivity (discounting N^{16} because of the 7.7 second half-life) and potential releases are limited to liquid releases of low activity water from the feedwater and condenser systems. Two other sources exist which contain radioactive species but in a form not amenable for release. The potential for accident releases from these two sources, the offgas system, and the condenser demineralizers, is reduced due to heavy shielding and compartmentalizing of the components.

12.2.1.4 Radwaste Building Source Terms

This section provides a summation of the significant radioactive source terms found in the ESBWR radwaste building. These source terms consist of those elements which are found to contain significant quantities of radioactive materials but do not include sources due to incidental contamination such as sources in valves due to deposition of corrosion or fission products species on the surfaces of the components.

Normal Operating Sources

Tables 12.2-13a through 12.2-13g and 12.2-14a through 12.2-14b provide source inventories for the major radwaste components for operation. These sources are based upon the stream concentrations given in Section 11.1 and represent sources for shielding calculations. These inventories should not be construed to represent sources for offsite release. A complete description of the ESBWR radwaste system is given in Sections 11.2 through 11.4.

Post-Accident Radioactive Sources

Potential releases in the radwaste building are contained by isolating the radwaste building atmosphere and sealing any water releases in the building. The radwaste building is seismically qualified and lined to prevent any potential water releases from high activity areas.

12.2.2 Airborne and Liquid Sources for Environmental Consideration

This subsection deals with the sources and parameters required to evaluate airborne and liquid releases during normal plant operation for compliance with 10 CFR 20 and 10 CFR 50, Appendix I criteria.

12.2.2.1 Airborne Sources

Airborne sources are calculated using the source terms given in Section 11.1. A ratio to an expected release rate is shown in Table 12.2-15 for average annual releases and subject to the criteria of Reference 12.2-1.

The bases for these calculations are shown in Table 12.2-15.

Airborne Source During Refueling

Airborne radioactivity during refueling is expected to be similar to that observed in operating sites. Experience has shown that airborne radioactivity can result from the water in the reactor cavity exceeding 38°C (100°F) and flaking of cobalt dioxide (CoO_2) from the steam dryer and separator if their surfaces are allowed to dry. Other potential airborne sources resulting from reactor vessel head and internals removal have been determined from experience. I^{131}, Co^{60}, Mn^{54}, Nb^{95}, Zr^{95}, Ru^{103}, and Ce^{144} were the major radioisotopes found with Ce^{141}, Cs^{137}, Co^{58}, and Cr^{51} at lower concentrations. The radioactive particulates ranged as high as 740 μBq/cm^3 (2 x 10^{-8} Ci/cm^3) and I^{131} as high as 1,500 μBq/cm^3 (4 x 10^{-8} μCi/cm^3).

To minimize airborne radioactivity the following actions are specified:

- Keep steam dryer and separator surfaces wet or covered.
- Cool fuel pools through large heat capacity heat exchangers.
- Fuel pool ventilation system designed to sweep the pool surface and prevent pool releases from mixing with the area atmosphere.

Annual Releases

Based upon the above criteria, the normal operating source terms are given in Table 12.2-16 and a comparison to 10 CFR 20 criteria is given in Table 12.2-17.

12.2.2.2 Airborne Dose Evaluation

Airborne doses were calculated based upon the criteria specified in Subsection 12.2.2.1 for compliance with 10 CFR 50, Appendix I. Doses were calculated using methodologies and conversion factors consistent with Regulatory Guides 1.109 and 1.111 as implemented in References 12.2-1 and 12.2-2. The results of this analysis are given in Table 12.2-18.

12.2.2.3 Liquid Sources

The ESBWR Radwaste System as described in Section 11.2 is designed to monitor and process all radioactive liquid streams in the ESBWR and to provide water management for those streams. Under normal conditions, the water management is not expected to result in any routine release of radioactive effluents in the liquid discharges. However, under some conditions such as high water inventory, some processed radioactive liquid effluents may be released. By administrative control, the discharge of these effluents through the discharge line is adjusted so that it can be shown that the discharge meets the requirements of 10 CFR 20 on isotopic concentration limits and Appendix I of 10 CFR 50 on annualized dose requirements.

A bounding annualized release is shown in Table 12.2-19. The plant configuration with respect to liquid discharges provides an additional minimum dilution factor of ten to the release prior to the point of closest exposure to the public. Such additional dilution factors may be justified on a site specific basis by the use of larger minimum discharge flow rates or dilution ponds or introducing the discharge pipe to a large flow discharge canal with mixing baffles to provide dilution prior to public exposure.

12.2.2.4 Liquid Doses

Liquid pathway doses were calculated based upon the criteria specified in Subsection 12.2.2.3 for compliance with 10 CFR 50, Appendix I. Dose conversion factors and methodologies consistent with Regulatory Guide 1.113 were used as described in Reference 12.2-4. The results of the dose calculation are given in Table 12.2-20.

12.2.3 COL Information

None.

12.2.4 References

12.2-1 U.S.N.R.C, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors," NUREG-0016, Revision 1, January 1979.

12.2-2 U.S.N.R.C., "GASPAR II Technical Reference and User Guide" NUREG/CR-4653, March 1987

12.2-3 U.S.N.R.C., "LADTAP II Technical Reference and User Guide" NUREG/CR-4013, April 1986

Table 12.2-1

Basic Reactor Data

A.	Reactor Thermal Power		4500 MW
B.	Average Power Density		54.33 kW/L
C.	Physical Dimensions		Fig. 12.2-1
	1.	Core Equivalent Radius, mm	2941.1
	2.	Inside Shroud Radius, mm	3171
	3.	Outside Shroud Radius, mm	3221
	4.	Inside Vessel Radius – Average, mm	3556
	5.	Outside Vessel Radius – Average, mm	3738
	6.	Outside Top Guide Radius, mm	3286
	7.	Vessel Top Head Inside Radius, mm	4866
	8.	Vessel Bottom Head Inside Radius, mm	4866
	9.	Bottom Head to Shell Knuckle Radius, mm	1092
		Elevation, mm	
	10.	Outside of Vessel Bottom Head	-263
	11.	Inside of Vessel Bottom Head	0
	12.	Intersect of Bottom Head Radius & Vessel Wall	2007
	13.	Bottom of Core Support Plate	4127.6
	14.	Top of Core Support Plate	4178.4
	15.	Bottom of Active Fuel	4405
	16.	Top of Active Fuel	7453
	17.	Bottom of Top Guide	7718.2
	18.	Top of Fuel Channel	7896.1
	19.	Normal Vessel Water Level	20720
	20.	Top of Steam Dryer	24775.5
	21.	Vessel Top Head Knuckle	25648
	22.	Inside of Vessel Top Head	27560
	23.	Outside of Vessel Top Head	27720

Table 12.2-1
Basic Reactor Data (Continued)

D. Material Densities* (g/cc)

Region	Coolant	UO_2	Zircaloy	316L Stainless
(A) Lower Plenum	0.768	0.000	0.000	0.178
(B) Core Plate & Beam	0.338	0.000	0.000	4.35
(C) Below Active Fuel	0.597	0.000	0.166	1.70
(D1) Peripheral Assemblies				
Plane 25	0.308	1.49	1.03	0.000
Plane 24	0.310	1.78	0.77	0.000
Plane 23	0.312	1.78	0.77	0.000
Plane 22	0.316	1.78	0.77	0.000
Plane 21	0.321	1.78	0.77	0.000
Plane 20	0.326	1.78	0.77	0.000
Plane 19	0.332	1.78	0.77	0.000
Plane 18	0.340	1.78	0.77	0.000
Plane 17	0.326	1.78	1.09	0.000
Plane 16	0.333	1.78	1.09	0.000
Plane 15	0.342	2.11	0.85	0.000
Plane 14	0.352	2.11	0.85	0.000
Plane 13	0.364	2.11	0.85	0.000
Plane 12	0.377	2.11	0.85	0.000
Plane 11	0.391	2.11	0.85	0.000
Plane 10	0.407	2.11	0.85	0.000
Plane 9	0.424	2.10	0.85	0.000
Plane 8	0.440	2.10	0.85	0.000
Plane 7	0.455	2.10	0.85	0.000

Table 12.2-1
Basic Reactor Data (Continued)

D. Material Densities* (g/cc)				
Region	**Coolant**	**UO_2**	**Zircaloy**	**316L Stainless**
Plane 6	0.467	2.10	0.85	0.000
Plane 5	0.474	2.10	0.85	0.000
Plane 4	0.478	2.10	0.85	0.000
Plane 3	0.480	2.10	0.85	0.000
Plane 2	0.482	2.10	0.85	0.000
Plane 1	0.483	2.17	0.85	0.000
Axial Avg	0.386	1.97	0.85	0.000
(D2) Interior Assemblies				
Plane 25	0.253	1.51	1.03	0.000
Plane 24	0.254	1.78	0.77	0.000
Plane 23	0.256	1.78	0.77	0.000
Plane 22	0.258	1.78	0.77	0.000
Plane 21	0.260	1.78	0.77	0.000
Plane 20	0.264	1.78	0.77	0.000
Plane 19	0.267	1.78	0.77	0.000
Plane 18	0.274	1.78	0.77	0.000
Plane 17	0.264	1.78	1.09	0.000
Plane 16	0.269	1.78	1.09	0.000
Plane 15	0.274	2.11	0.85	0.000
Plane 14	0.280	2.11	0.85	0.000
Plane 13	0.286	2.11	0.85	0.000
Plane 12	0.295	2.11	0.85	0.000
Plane 11	0.304	2.11	0.85	0.000
Plane 10	0.315	2.11	0.85	0.000
Plane 9	0.329	2.10	0.85	0.000
Plane 8	0.345	2.10	0.85	0.000
Plane 7	0.364	2.10	0.85	0.000
Plane 6	0.386	2.10	0.85	0.000
Plane 5	0.412	2.10	0.85	0.000

Table 12.2-1
Basic Reactor Data (Continued)

D. Material Densities* (g/cc)

Region	Coolant	UO_2	Zircaloy	316L Stainless
Plane 4	0.441	2.10	0.85	0.000
Plane 3	0.466	2.10	0.85	0.000
Plane 2	0.479	2.10	0.85	0.000
Plane 1	0.483	2.17	0.85	0.000
Axial Avg	0.323	1.97	0.85	0.000
(E) Bypass Water	0.735	0.000	0.000	0.000
(F) Above Active Fuel	0.234	0.000	1.10	0.255
(G) Top Guide	0.240	0.000	1.00	1.21
(H) Chimney	0.390	0.000	0.000	0.000
(J,K) Downcomer	0.768	0.000	0.000	0.000

* See Figure 12.2-1 for location schematic.

Table 12.2-1
Basic Reactor Data (Continued)

E. Equilibrium Cycle Relative Power Distribution
Two-Dimensional Distribution at Core Midplane

Node	1	2	3	4	5	6	7	8	9	10
1										
2										
3										
4									0.51	0.65
5								0.58	0.82	0.97
6							0.58	0.85	1.04	1.14
7						0.58	0.73	1.06	1.13	1.26
8					0.58	0.85	1.06	1.19	1.31	1.22
9				0.51	0.82	1.04	1.13	1.31	1.32	1.43
10				0.65	0.97	1.14	1.26	1.22	1.43	1.30
11			0.54	0.87	1.09	1.23	1.06	1.26	1.43	1.45
12			0.65	1.01	1.21	1.22	1.23	1.15	1.47	1.33
13		0.47	0.86	1.13	1.24	1.41	1.44	1.49	1.42	1.52
14		0.61	0.99	1.21	1.36	1.30	1.51	1.49	1.56	1.38
15	0.44	0.70	1.04	1.16	1.40	1.47	1.32	1.48	1.42	1.53
16	0.51	0.90	1.06	1.25	1.42	1.32	1.41	1.47	1.57	1.53
17	0.56	0.94	1.12	1.33	1.20	1.47	1.36	1.54	1.40	1.54
18	0.58	0.94	1.17	1.31	1.42	1.31	1.48	1.37	1.52	1.36
19	0.55	0.92	0.93	1.21	1.27	1.44	1.24	1.26	1.35	1.45

Table 12.2-1
Basic Reactor Data (Continued)

E. Equilibrium Cycle Relative Power Distribution (Cont.)
Two-Dimensional Distribution at Core Midplane

Node	11	12	13	14	15	16	17	18	19
1					0.44	0.51	0.56	0.58	0.55
2			0.47	0.61	0.70	0.90	0.94	0.94	0.92
3	0.54	0.65	0.86	0.99	1.04	1.06	1.12	1.17	0.93
4	0.87	1.01	1.13	1.21	1.16	1.25	1.33	1.31	1.21
5	1.09	1.21	1.24	1.36	1.40	1.42	1.20	1.42	1.27
6	1.23	1.22	1.41	1.30	1.47	1.32	1.47	1.31	1.44
7	1.06	1.23	1.44	1.51	1.32	1.41	1.36	1.48	1.24
8	1.26	1.15	1.49	1.49	1.48	1.47	1.54	1.37	1.26
9	1.43	1.47	1.42	1.56	1.42	1.57	1.40	1.52	1.35
10	1.45	1.33	1.52	1.38	1.53	1.53	1.54	1.36	1.45
11	1.17	1.25	1.36	1.51	1.22	1.28	1.35	1.46	1.09
12	1.25	1.23	1.50	1.50	1.28	1.26	1.49	1.44	1.10
13	1.36	1.50	1.36	1.52	1.35	1.49	1.35	1.48	1.31
14	1.51	1.50	1.52	1.35	1.47	1.33	1.50	1.35	1.46
15	1.22	1.28	1.35	1.47	1.15	1.20	1.46	1.47	1.19
16	1.28	1.26	1.49	1.33	1.20	1.20	1.46	1.36	1.23
17	1.35	1.49	1.35	1.50	1.46	1.46	1.32	1.47	1.30
18	1.46	1.44	1.48	1.35	1.47	1.36	1.47	1.30	1.40
19	1.09	1.10	1.31	1.46	1.19	1.23	1.30	1.40	1.06

Table 12.2-1
Basic Reactor Data (Continued)

F. End of Equilibrium Cycle Average Exposure Distribution
Bundle Average Exposure (MWd/StU)

Node	1	2	3	4	5	6	7	8	9	10
1										
2										
3										
4									42135	42200
5								44916	24660	13536
6							42187	26288	14627	16072
7						42187	44815	14800	32578	17936
8					44916	26288	14800	16864	18611	37338
9				42135	24660	14627	32578	18611	35229	19936
10				42200	13536	16072	17936	37338	19936	39140
11			42236	27737	15354	17369	44183	33665	19870	19946
12			41974	13968	16968	36545	35724	44901	20370	38251
13		45216	29357	15688	34026	19729	19933	20644	38763	20957
14		41742	13389	16683	18918	38647	20493	33781	21393	39583
15	39035	42286	33302	36166	19141	19854	41822	33313	39236	20969
16	42146	29095	34717	35216	19393	39063	37227	33683	21197	20859
17	41231	28516	33232	18158	45645	20062	38553	20869	39248	21126
18	39965	27820	15960	17998	19523	38907	20278	38586	20929	38840
19	40864	12094	41543	28901	37990	19733	38479	38338	38535	20013

Table 12.2-1
Basic Reactor Data (Continued)

F. End of Equilibrium Cycle Average Exposure Distribution (Cont.)
Bundle Average Exposure (MWd/StU)

Node	11	12	13	14	15	16	17	18	19
1					39035	42146	41231	39965	40864
2			45216	41742	42286	29095	28516	27820	12094
3	42236	41974	29357	13389	33302	34717	33232	15960	41543
4	27737	13968	15688	16683	36166	35216	18158	17998	28901
5	15354	16968	34026	18918	19141	19393	45645	19523	37990
6	17369	36545	19729	38647	19854	39063	20062	38907	19733
7	44183	35724	19933	20493	41822	37227	38553	20278	38479
8	33665	44901	20644	33781	33313	33683	20869	38586	38338
9	19870	20370	38763	21393	39236	21197	39248	20929	38535
10	19946	38251	20957	39583	20969	20859	21126	38840	20013
11	43322	36545	38199	20683	44544	37649	38963	20135	36588
12	36545	37338	20460	20500	37348	37800	20495	19905	35631
13	38199	20460	39755	20863	38755	20465	39197	20501	38429
14	20683	20500	20863	39511	20109	38489	20602	38692	20003
15	44544	37348	38755	20109	43177	36882	19949	20152	42689
16	37649	37800	20465	38489	36882	37064	20028	38502	37701
17	38963	20495	39197	20602	19949	20028	39261	20324	38629
18	20135	19905	20501	38692	20152	38502	20324	39137	19492
19	36588	35631	38429	20003	42689	37701	38629	19492	36118

Table 12.2-1
Basic Reactor Data (Continued)

G. Axial Power Distribution

Node	Node Mid-Point Elevation (mm above BAF)	Relative Power
25	2987.0	0.19
24	2865.1	0.38
23	2743.2	0.53
22	2621.3	0.66
21	2499.4	0.77
20	2377.4	0.86
19	2255.5	0.93
18	2133.6	0.97
17	2011.7	1.00
16	1889.8	1.01
15	1767.8	1.16
14	1645.9	1.19
13	1524.0	1.22
12	1402.1	1.24
11	1280.2	1.26
10	1158.2	1.27
9	1036.3	1.28
8	914.4	1.28
7	792.5	1.29
6	670.6	1.30
5	548.6	1.29
4	426.7	1.26
3	304.8	1.17
2	182.9	0.96
1	61.0	0.52

Table 12.2-2

Neutron Fluxes at Core Boundary and RPV

Part A. Neutron Spectrum at Peak Elevation

Upper Energy (eV)	Core Boundary (neutrons/cm^2-sec)	RPV Inside Surface (neutrons/cm^2-sec)	RPV 1/4T Thickness (neutrons/cm^2-sec)
2.000E+7	2.7E+10	1.1E+08	5.9E+07
1.000E+7	4.8E+11	9.9E+08	5.1E+08
6.065E+6	2.0E+12	2.0E+09	1.0E+09
3.679E+6	4.0E+12	2.5E+09	1.4E+09
2.231E+6	4.3E+12	2.8E+09	2.1E+09
1.353E+6	3.8E+12	2.8E+09	2.6E+09
8.209E+5	3.8E+12	3.1E+09	3.4E+09
4.979E+5	2.6E+12	2.3E+09	2.7E+09
3.020E+5	2.2E+12	1.6E+09	1.7E+09
1.832E+5	3.2E+12	2.3E+09	2.4E+09
6.738E+4	2.4E+12	1.4E+09	1.2E+09
2.479E+4	2.1E+12	1.2E+09	9.6E+08
9.119E+3	2.0E+12	1.1E+09	7.4E+08
3.355E+3	2.0E+12	1.1E+09	7.4E+08
1.234E+3	1.9E+12	1.1E+09	6.9E+08
4.540E+2	2.0E+12	1.1E+09	6.9E+08
1.670E+2	2.0E+12	1.1E+09	7.8E+08
6.144E+1	1.8E+12	9.2E+08	4.3E+08
2.260E+1	8.4E+11	4.8E+08	2.3E+08
1.371E+1	8.4E+11	4.8E+08	2.3E+08
8.315E+0	7.4E+11	4.7E+08	2.2E+08
5.044E+0	7.8E+11	4.6E+08	2.1E+08
3.059E+0	1.5E+12	8.7E+08	3.3E+08
1.125E+0	1.4E+12	7.7E+08	2.3E+08
4.140E-1	1.1E+12	6.5E+08	1.3E+08
1.523E-1	1.5E+13	5.1E+10	3.3E+09
1.389E-4			
Total Flux	6.5E+13	8.5E+10	2.9E+10
Fast Flux (E>1 MeV)	1.3E+13	1.0E+10	6.6E+09

Table 12.2-2
Neutron Fluxes at Core Boundary and RPV (Continued)

Part B. Relative Fast Flux (E>1 MeV)

Distance from Bottom of Active Fuel (mm)	Relative Flux at RPV Inside Surface
3048.0 (TAF)	0.241
2987.0	0.301
2865.1	0.443
2743.2	0.590
2621.3	0.720
2499.4	0.822
2377.4	0.892
2255.5	0.937
2133.6	0.959
2011.7	0.974
1889.8	0.985
1767.8	0.995
1645.9	1.000
1524.0	0.996
1402.1	0.980
1280.2	0.952
1158.2	0.917
1036.3	0.873
914.4	0.823
792.5	0.767
670.6	0.706
548.6	0.640
426.7	0.567
304.8	0.481
182.9	0.380
61.0	0.269
0.0 (BAF)	0.218

Table 12.2-3

Gamma Ray Source Energy Spectra

A. Gamma Ray Sources During Operation (MeV/sec-cm^3)

Upper Energy (MeV)	Core	Bypass Water	Shroud	Downcomer	RPV
30	0.0E+00	0.0E+00	3.7E+04	0.0E+00	1.8E+02
17	2.8E+02	7.7E+01	6.3E+05	5.5E+00	1.7E+03
12	8.8E+04	1.9E+03	8.1E+07	1.2E+02	5.1E+05
10	3.0E+05	1.3E+04	1.4E+11	6.9E+02	3.2E+08
9	3.4E+08	8.9E+06	4.6E+11	3.8E+05	3.4E+08
8	2.5E+10	2.0E+05	1.0E+12	7.1E+03	3.6E+09
7	1.1E+11	6.8E+05	2.2E+11	2.2E+04	5.8E+08
6	4.3E+11	2.4E+06	2.4E+11	6.0E+04	6.9E+08
5	1.3E+12	1.2E+07	1.7E+11	4.6E+05	5.7E+08
4	2.6E+12	6.1E+08	1.8E+11	2.1E+07	6.2E+08
3	2.5E+12	4.3E+07	8.8E+10	1.5E+06	3.5E+08
2.5	3.9E+12	7.0E+11	1.4E+11	1.6E+10	5.0E+08
2	3.0E+12	0.0E+00	1.6E+11	0.0E+00	6.1E+08
1.66	3.4E+12	0.0E+00	1.4E+11	0.0E+00	1.9E+08
1.33	3.4E+12	8.3E+07	2.7E+10	1.9E+06	1.1E+08
1	3.5E+12	7.3E+07	2.4E+11	1.7E+06	1.0E+09
0.75	3.0E+12	3.4E+05	1.7E+10	1.2E+04	4.5E+07
0.525	3.2E+11	0.0E+00	9.9E+08	0.0E+00	1.7E+06
0.5	1.7E+12	1.2E+05	1.2E+10	5.6E+03	4.2E+07
0.3	6.9E+11	9.3E+05	1.2E+09	4.3E+04	2.6E+06
0.2	5.2E+11	2.1E+06	1.6E+09	7.6E+04	6.4E+06
0.1	1.3E+11	0.0E+00	6.9E+07	0.0E+00	1.2E+05
0.06	7.4E+10	0.0E+00	1.9E+09	0.0E+00	7.9E+06
0.03	2.5E+10	0.0E+00	7.8E+08	0.0E+00	3.5E+06
0.01					
Total	3.1E+13	7.0E+11	3.3E+12	1.7E+10	9.6E+09

Table 12.2-3
Gamma Ray Source Energy Spectra (Continued)

B. Post-Operation Gamma Sources in Core (MeV/sec-MWt)

Upper Energy (MeV)	Time after Shutdown 0 Sec	1 Day	4 Days	1 Month
11	9.1E+09	1.0E+05	1.0E+05	9.7E+04
8	2.7E+13	6.7E+05	6.6E+05	6.2E+05
6	2.2E+15	2.5E+10	4.2E+06	3.9E+06
4	3.2E+15	2.5E+12	2.1E+12	5.7E+11
3	4.7E+15	2.2E+14	2.0E+14	4.9E+13
2.5	9.1E+15	2.3E+14	1.7E+14	5.9E+13
2	1.4E+16	3.9E+15	3.3E+15	8.3E+14
1.5	2.7E+16	1.5E+15	7.0E+14	1.9E+14
1	3.2E+16	7.8E+15	5.7E+15	3.5E+15
0.7	2.0E+16	5.5E+15	3.2E+15	1.3E+15
0.45	8.3E+15	1.2E+15	7.6E+14	1.6E+14
0.3	8.8E+15	2.4E+15	1.1E+15	1.1E+14
0.15	3.3E+15	1.7E+15	8.6E+14	1.5E+14
0.1	2.5E+15	4.8E+14	2.7E+14	4.7E+13
0.07	1.1E+15	1.8E+14	1.1E+14	4.3E+13
0.045	7.3E+14	1.8E+14	1.1E+14	3.8E+13
0.03	5.7E+14	9.9E+13	6.3E+13	2.3E+13
0.02	1.2E+15	2.9E+14	1.5E+14	3.7E+13
0				
Total	1.4E+17	2.6E+16	1.7E+16	6.5E+15

Table 12.2-4

Neutron and Gamma Ray Fluxes Outside the Vessel Wall

Neutron		Gamma Ray	
Upper Energy (eV)	Neutron Flux (neutrons/cm^2-sec)	Upper Energy (MeV)	Gamma Ray Energy Flux (MeV/cm^2-sec)
2.000E+7	8.0E+06	30	3.3E+03
1.000E+7	5.8E+07	17	2.9E+04
6.065E+6	9.8E+07	12	2.2E+06
3.679E+6	1.7E+08	10	5.7E+08
2.231E+6	3.3E+08	9	1.5E+09
1.353E+6	6.3E+08	8	5.3E+09
8.209E+5	1.0E+09	7	2.1E+09
4.979E+5	8.8E+08	6	2.3E+09
3.020E+5	5.2E+08	5	2.5E+09
1.832E+5	7.2E+08	4	2.8E+09
6.738E+4	2.4E+08	3	1.5E+09
2.479E+4	2.0E+08	2.5	2.2E+09
9.119E+3	9.7E+07	2	1.5E+09
3.355E+3	8.2E+07	1.66	1.3E+09
1.234E+3	6.9E+07	1.33	1.3E+09
4.540E+2	5.9E+07	1	1.5E+09
1.670E+2	6.9E+07	0.75	7.4E+08
6.144E+1	3.0E+07	0.525	1.0E+08
2.260E+1	1.4E+07	0.5	8.5E+08
1.371E+1	1.3E+07	0.3	4.8E+08
8.315E+0	1.2E+07	0.2	2.9E+08
5.044E+0	1.0E+07	0.1	3.3E+07
3.059E+0	1.4E+07	0.06	8.4E+05
1.125E+0	7.0E+06	0.03	6.0E+03
4.140E-1	2.6E+06	0.01	
1.523E-1	1.2E+06	Total	2.9E+10
1.389E-4			
Total	5.3E+09		

Table 12.2-5

Radioactive Sources in the Control Rod Drive System

Control Rod Drive Radiation Survey Data		
	Gamma Dose Measured at Contact, mSv/hr	
Component	Before Cleaning	After Cleaning
Seal Housing (Spool Piece)	1.5E-01	0.0E+00
Rotating Ball Spindle	0.0E+00	3.0E-01
Hollow Piston	7.5E-01	3.8E-01
Throttle Bushing	6.0E-01	6.0E-01
Guide Tube	4.5E-01	3.0E-01
Motor/Synchro Assembly	3.0E-02	<1.5E-02
Cylinder Tube/Flange	3.3E+00	3.0E-01

Control Blade Principal Isotopes	
Isotope	**MBq/Blade**
Cr-51	5.2E+09
Mn-54	3.4E+08
Fe-55	5.9E+09
Co-58m	3.3E+08
Co-60	4.1E+09
Ni-63	1.9E+08
Total	1.6E+10

Table 12.2-6a
RWCU/SDC Heat Exchanger
Regenerative Heat Exchanger Tube Sides

Class	Isotope	MBq	Ci	Class	Isotope	MBq	Ci
Class 2	I-131	2.83E+02	7.64E-03	Class 6	Sr-92	1.17E+03	3.16E-02
	I-132	2.61E+03	7.05E-02		Y-91	5.13E+00	1.39E-04
	I-133	1.92E+03	5.20E-02		Y-91m	2.30E+01	6.21E-04
	I-134	4.91E+03	1.33E-01		Y-92	7.08E+02	1.91E-02
	I-135	2.86E+03	7.72E-02		Y-93	4.79E+02	1.30E-02
					Zr-95	1.04E+00	2.80E-05
Class 3	Rb-89	4.53E+02	1.22E-02		Nb-95	1.04E+00	2.80E-05
	Cs-134	3.40E+00	9.19E-05		Mo-99	2.46E+02	6.65E-03
	Cs-136	2.23E+00	6.02E-05		Tc-99m	2.46E+02	6.65E-03
	Cs-137	8.68E+00	2.35E-04		Ru-103	2.53E+00	6.84E-05
	Cs-138	9.67E+02	2.61E-02		Rh-103m	2.53E+00	6.84E-05
					Ru-106	3.95E-01	1.07E-05
Class 5	H-3	2.92E+02	7.88E-03		Rh-106	3.95E-01	1.07E-05
					Ag-110m	1.24E-01	3.36E-06
Class 6	Na-24	2.41E+02	6.50E-03		Te-129	2.30E-01	6.21E-06
	P-32	5.18E+00	1.40E-04		Te-129m	5.13E+00	1.39E-04
	Cr-51	3.74E+02	1.01E-02		Te-131	4.74E-01	1.28E-05
	Mn-54	4.41E+00	1.19E-04		Te-131m	1.23E+01	3.33E-04
	Mn-56	2.64E+03	7.12E-02		Te-132	1.23E+00	3.34E-05
	Fe-55	1.24E+02	3.36E-03		Ba-136m	6.92E-01	1.87E-05
	Fe-59	3.79E+00	1.02E-04		Ba-137m	5.01E+00	1.35E-04
	Co-58	1.28E+01	3.46E-04		Ba-140	4.95E+01	1.34E-03
	Co-60	2.51E+01	6.79E-04		La-140	4.95E+01	1.34E-03
	Ni-63	1.24E-01	3.36E-06		Ce-141	3.95E+00	1.07E-04
	Cu-64	3.70E+02	1.00E-02		Ce-144	3.95E-01	1.07E-05
	Zn-65	1.28E+02	3.46E-03		Pr-144	3.95E-01	1.07E-05
	Sr-89	1.25E+01	3.39E-04		W-187	3.74E+01	1.01E-03
	Sr-90	9.02E-01	2.44E-05		Np-239	9.76E+02	2.64E-02
	Y-90	9.02E-01	2.44E-05				
	Sr-91	4.85E+02	1.31E-02		**Total**	**2.28E+04**	**6.15E-01**

Table 12.2-6b

RWCU/SDC Heat Exchanger

Non-Regenerative Heat Exchanger Tube Sides

Class	Isotope	MBq	Ci	Class	Isotope	MBq	Ci
Class 2	I-131	2.16E+03	5.85E-02	Class 6	Sr-92	8.77E+03	2.37E-01
	I-132	2.02E+04	5.46E-01		Y-91	3.93E+01	1.06E-03
	I-133	1.56E+04	4.21E-01		Y-91m	2.15E+02	5.81E-03
	I-134	3.61E+04	9.77E-01		Y-92	5.43E+03	1.47E-01
	I-135	2.10E+04	5.66E-01		Y-93	3.60E+03	9.72E-02
					Zr-95	7.72E+00	2.09E-04
Class 3	Rb-89	3.23E+03	8.74E-02		Nb-95	7.72E+00	2.09E-04
	Cs-134	2.55E+01	6.89E-04		Mo-99	1.76E+03	4.75E-02
	Cs-136	1.65E+01	4.45E-04		Tc-99m	1.76E+03	4.75E-02
	Cs-137	6.84E+01	1.85E-03		Ru-103	1.86E+01	5.02E-04
	Cs-138	7.48E+03	2.02E-01		Rh-103m	1.86E+01	5.02E-04
					Ru-106	2.92E+00	7.88E-05
Class 5	H-3	2.15E+03	5.82E-02		Rh-106	2.92E+00	7.88E-05
					Ag-110m	9.59E-01	2.59E-05
Class 6	Na-24	1.85E+03	5.01E-02		Te-129	2.15E+00	5.81E-05
	P-32	3.83E+01	1.03E-03		Te-129m	3.93E+01	1.06E-03
	Cr-51	2.90E+03	7.84E-02		Te-131	4.37E+00	1.18E-04
	Mn-54	3.34E+01	9.02E-04		Te-131m	9.57E+01	2.59E-03
	Mn-56	2.02E+04	5.45E-01		Te-132	9.87E+00	2.67E-04
	Fe-55	9.59E+02	2.59E-02		Ba-136m	5.10E+00	1.38E-04
	Fe-59	2.90E+01	7.84E-04		Ba-137m	4.34E+01	1.17E-03
	Co-58	9.45E+01	2.55E-03		Ba-140	3.76E+02	1.02E-02
	Co-60	1.94E+02	5.25E-03		La-140	3.76E+02	1.02E-02
	Ni-63	9.59E-01	2.59E-05		Ce-141	2.92E+01	7.88E-04
	Cu-64	2.81E+03	7.60E-02		Ce-144	2.92E+00	7.88E-05
	Zn-65	3.11E+02	8.41E-03		Pr-144	2.92E+00	7.88E-05
	Sr-89	9.48E+01	2.56E-03		W-187	2.89E+02	7.82E-03
	Sr-90	7.11E+00	1.92E-04		Np-239	7.67E+03	2.07E-01
	Y-90	7.11E+00	1.92E-04				
	Sr-91	3.64E+03	9.84E-02		**Totals**	**1.72E+05**	**4.64E+00**

Table 12.2-6c
RWCU/SDC Heat Exchanger
Regenerative Heat Exchanger Shell Side

Class	Isotope	MBq	Ci	Class	Isotope	MBq	Ci
Class 2	I-131	3.49E+01	9.44E-04	Class 6	Sr-92	1.36E+02	3.69E-03
	I-132	3.04E+02	8.22E-03		Y-91	6.16E-01	1.66E-05
	I-133	2.45E+02	6.62E-03		Y-91m	4.01E+00	1.08E-04
	I-134	5.68E+02	1.54E-02		Y-92	8.50E+01	2.30E-03
	I-135	3.24E+02	8.75E-03		Y-93	5.59E+01	1.51E-03
					Zr-95	1.23E-01	3.32E-06
Class 3	Rb-89	2.52E+02	6.82E-03		Nb-95	1.23E-01	3.32E-06
	Cs-134	2.04E+00	5.51E-05		Mo-99	2.81E+01	7.60E-04
	Cs-136	1.27E+00	3.43E-05		Tc-99m	2.81E+01	7.60E-04
	Cs-137	5.17E+00	1.40E-04		Ru-103	3.04E-01	8.21E-06
	Cs-138	5.61E+02	1.52E-02		Rh-103m	3.04E-01	8.21E-06
					Ru-106	4.63E-02	1.25E-06
Class 5	H-3	3.37E+02	9.10E-03		Rh-106	4.63E-02	1.25E-06
					Ag-110m	1.47E-02	3.97E-07
Class 6	Na-24	2.76E+01	7.45E-04		Te-129	4.01E-02	1.08E-06
	P-32	6.08E-01	1.64E-05		Te-129m	5.98E-01	1.62E-05
	Cr-51	4.35E+01	1.18E-03		Te-131	8.38E-02	2.27E-06
	Mn-54	5.26E-01	1.42E-05		Te-131m	1.44E+00	3.88E-05
	Mn-56	3.18E+02	8.59E-03		Te-132	1.50E-01	4.05E-06
	Fe-55	1.47E+01	3.97E-04		Ba-136m	8.02E-02	2.17E-06
	Fe-59	4.57E-01	1.24E-05		Ba-137m	7.34E-01	1.98E-05
	Co-58	1.50E+00	4.06E-05		Ba-140	5.80E+00	1.57E-04
	Co-60	2.97E+01	8.03E-04		La-140	5.80E+00	1.57E-04
	Ni-63	1.47E-02	3.97E-07		Ce-141	4.63E-01	1.25E-05
	Cu-64	4.34E+01	1.17E-03		Ce-144	4.63E-02	1.25E-06
	Zn-65	1.50E+01	4.06E-04		Pr-144	4.46E-02	1.21E-06
	Sr-89	1.44E+00	3.89E-05		W-187	4.46E+00	1.21E-04
	Sr-90	1.02E-01	2.77E-06		Np-239	1.17E+02	3.15E-03
	Y-90	1.02E-01	2.77E-06				
	Sr-91	5.66E+01	1.53E-03		**Totals**	**3.66E+03**	**9.90E-02**

Table 12.2-7

RWCU Demineralizer

Total Activity 3.45E+08 MBq
9.31E+09 μCi

Halogens			Soluble Fission Products			Insoluble Fission Products			Activation Products		
Isotope	MBq	μCi	Isotope	MBq	μCi	Isotope	MBq	μCi	Isotope	MBq	μCi
I-131	9.48E+06	2.56E+08	Rb-89	1.75E+04	4.72E+05	Y-91	1.88E+06	5.08E+07	Na-24	6.02E+05	1.63E+07
I-132	1.02E+06	2.76E+07	Sr-89	2.54E+06	6.86E+07	Y-91m	7.65E+05	2.07E+07	P-32	2.93E+05	7.91E+06
I-133	6.99E+06	1.89E+08	Sr-90	8.85E+05	2.39E+07	Y-92	9.21E+05	2.49E+07	Cr-51	4.18E+07	1.13E+09
I-134	6.97E+05	1.88E+07	Y-90	8.85E+05	2.39E+07	Y-93	8.19E+05	2.21E+07	Mn-54	3.11E+06	8.41E+07
I-135	3.05E+06	8.24E+07	Sr-91	7.48E+05	2.02E+07	Zr-95	2.63E+05	7.12E+06	Mn-56	1.16E+06	3.14E+07
			Sr-92	5.12E+05	1.38E+07	Nb-95	4.04E+05	1.09E+07	Fe-55	1.13E+08	3.05E+09
			Mo-99	2.64E+06	7.12E+07	Nb-95m	4.74E+03	1.28E+05	Fe-59	6.69E+05	1.81E+07
			Tc-99m	2.64E+06	7.12E+07	Ru-103	4.05E+05	1.09E+07	Co-58	3.45E+06	9.31E+07
			Te-129	9.11E+05	2.46E+07	Rh-103m	4.05E+05	1.09E+07	Co-60	2.46E+07	6.64E+08
			Te-129m	6.84E+05	1.85E+07	Ru-106	2.74E+05	7.42E+06	Ni-63	1.24E+05	3.36E+06
			Te-131	2.70E+04	7.29E+05	Rh-106	2.74E+05	7.42E+06	Cu-64	7.79E+05	2.11E+07
			Te-131m	6.26E+04	1.69E+06	La-140	2.90E+06	7.84E+07	Zn-65	7.78E+07	2.10E+09
			Te-132	1.64E+04	4.43E+05	Ce-141	4.97E+05	1.34E+07	Ag-110m	8.13E+04	2.20E+06
			Cs-134	2.89E+06	7.81E+07	Ce-144	2.74E+05	7.42E+06	W-187	1.45E+05	3.91E+06
			Cs-136	1.11E+05	3.01E+06	Pr-144	2.57E+05	6.95E+06			
			Cs-137	8.84E+06	2.39E+08						
			Cs-138	8.42E+04	2.28E+06						
			Ba-136m	3.57E+04	9.65E+05						
			Ba-137m	8.34E+06	2.25E+08						
			Ba-140	2.56E+06	6.92E+07						
			Np-239	9.23E+06	2.50E+08						
TOTAL	**2.12E+07**	**5.74E+08**	**TOTAL**	**4.47E+07**	**1.21E+09**	**TOTAL**	**1.12E+07**	**3.03E+08**	**TOTAL**	**2.67E+08**	**7.23E+09**

Table 12.2-8
FAPCS Filter Demineralizer

Total Activity	2.74E+07	MBq
	7.40E+08	µCi

Halogens			Soluble Fission Products			Insoluble Fission Products			Activation Products		
Isotope	MBq	µCi	Isotope	MBq	µCi	Isotope	MBq	µCi	Isotope	MBq	µCi
I-131	6.99E+05	1.89E+07	Sr-89	2.20E+05	5.95E+06	Y-91	1.56E+05	4.21E+06	Na-24	5.77E+03	1.56E+05
I-132	1.20E+03	3.23E+04	Sr-90	7.81E+04	2.11E+06	Y-91m	2.22E+03	6.01E+04	P-32	2.25E+04	6.08E+05
I-133	1.26E+05	3.40E+06	Y-90	7.81E+04	2.11E+06	Y-92	1.79E+01	4.85E+02	Cr-51	3.46E+06	9.34E+07
I-135	1.79E+03	4.84E+04	Sr-91	1.98E+03	5.36E+04	Y-93	2.80E+03	7.56E+04	Mn-54	2.66E+05	7.19E+06
			Sr-92	2.12E-01	5.73E+00	Zr-95	2.28E+03	6.17E+04	Fe-55	9.82E+06	2.65E+08
			Mo-99	1.41E+05	3.80E+06	Nb-95	3.34E+04	9.01E+05	Fe-59	5.69E+04	1.54E+06
			Tc-99m	1.34E+05	3.61E+06	Nb-95m	4.37E+02	1.18E+04	Co-58	2.95E+05	7.96E+06
			Te-129	7.65E+04	2.07E+06	Ru-103	3.38E+04	9.12E+05	Co-60	2.11E+06	5.71E+07
			Te-129m	5.81E+04	1.57E+06	Rh-103m	3.38E+04	9.12E+05	Ni-63	1.12E+04	3.02E+05
			Te-131	8.02E+02	2.17E+04	Ru-106	2.40E+04	6.49E+05	Cu-64	4.98E+03	1.35E+05
			Te-131m	1.78E+03	4.82E+04	Rh-106	2.40E+04	6.49E+05	Zn-65	6.67E+06	1.80E+08
			Te-132	9.34E+02	2.53E+04	La-140	2.22E+05	6.00E+06	Ag-110	3.43E+01	9.26E+02
			Cs-134	2.55E+05	6.89E+06	Ce-141	4.12E+04	1.11E+06	Ag-110m	7.11E+03	1.92E+05
			Cs-136	8.74E+03	2.36E+05	Ce-144	2.23E+04	6.03E+05	W-187	3.26E+03	8.80E+04
			Cs-137	7.68E+05	2.07E+07	Pr-144	2.23E+04	6.03E+05			
			Ba-136m	2.81E+03	7.58E+04	Pr-144m	7.29E+02	1.97E+04			
			Ba-137m	7.18E+05	1.94E+07						
			Ba-140	2.05E+05	5.54E+06						
			U-235m	4.74E+01	1.28E+03						
			Np-239	4.53E+05	1.22E+07						
			Pu-239	4.74E+01	1.28E+03						
TOTAL	**8.28E+05**	**2.24E+07**	**TOTAL**	**3.20E+06**	**8.65E+07**	**TOTAL**	**6.21E+05**	**1.68E+07**	**TOTAL**	**2.27E+07**	**6.14E+08**

Table 12.2-9

FAPCS Backwash Receiving Tank

Total Activity	9.18E+06	MBq
	2.48E+08	µCi

Halogens			Soluble Fission Products			Insoluble Fission Products			Activation Products		
Isotope	MBq	µCi	Isotope	MBq	µCi	Isotope	MBq	µCi	Isotope	MBq	µCi
I-131	2.33E+05	6.29E+06	Sr-89	7.11E+04	1.92E+06	Y-91	5.13E+04	1.39E+06	Na-24	1.96E+03	5.28E+04
I-132	3.91E+02	1.06E+04	Sr-90	2.60E+04	7.03E+05	Y-91m	7.29E+02	1.97E+04	P-32	7.65E+03	2.07E+05
I-133	4.20E+04	1.13E+06	Y-90	2.60E+04	7.03E+05	Y-92	5.90E+00	1.60E+02	Cr-51	1.17E+06	3.16E+07
I-135	5.90E+02	1.60E+04	Sr-91	6.67E+02	1.80E+04	Y-93	9.39E+02	2.54E+04	Mn-54	9.05E+04	2.45E+06
			Sr-92	7.06E-02	1.91E+00	Zr-95	7.37E+02	1.99E+04	Fe-55	3.27E+06	8.85E+07
			Mo-99	4.75E+04	1.28E+06	Nb-96	2.37E+04	6.40E+05	Fe-59	1.90E+04	5.12E+05
			Tc-99m	4.39E+04	1.19E+06	Nb-95m	1.38E+02	3.74E+03	Co-58	1.00E+05	2.70E+06
			Te-129	2.59E+04	6.99E+05	Ru-103	1.11E+04	3.01E+05	Co-60	6.86E+05	1.85E+07
			Te-129m	1.88E+04	5.08E+05	Rh-103m	1.11E+04	3.01E+05	Ni-63	3.72E+03	1.01E+05
			Te-131	2.62E+02	7.09E+03	Ru-106	8.23E+03	2.23E+05	Cu-64	1.66E+03	4.49E+04
			Te-131m	5.89E+02	1.59E+04	Rh-106	8.23E+03	2.23E+05	Zn-65	2.28E+06	6.16E+07
			Te-132	3.17E+02	8.58E+03	La-140	7.51E+04	2.03E+06	Ag-110	1.13E+01	3.05E+02
			Cs-134	8.50E+04	2.30E+06	Ce-141	1.41E+04	3.80E+05	Ag-110m	2.37E+03	6.41E+04
			Cs-136	2.91E+03	7.88E+04	Ce-144	7.55E+03	2.04E+05	W-187	1.09E+03	2.93E+04
			Cs-137	2.50E+05	6.76E+06	Pr-144	7.55E+03	2.04E+05			
			Ba-136m	9.11E+02	2.46E+04	Pr-144m	2.37E+02	6.40E+03			
			Ba-137m	2.34E+05	6.31E+06						
			Ba-140	6.49E+04	1.75E+06						
			U-235m	1.60E+01	4.33E+02						
			Np-239	1.50E+05	4.05E+06						
			Pu-239	1.60E+01	4.33E+02						
TOTAL	**2.76E+05**	**7.46E+06**	**TOTAL**	**1.05E+06**	**2.83E+07**	**TOTAL**	**2.21E+05**	**5.97E+06**	**TOTAL**	**7.64E+06**	**2.06E+08**

Table 12.2-10a
Offgas System
Steam Jet Air Ejector Inventory in MBq

Isotope	1st Ataje (MBq)	Intercooler Condenser (MBq)	2nd Ataje Ejector (MBq)
Kr-83m	1.20E+01	3.60E+02	3.60E+01
Kr-85m	2.10E+01	6.30E+02	6.30E+01
Kr-85	6.60E-02	1.98E+00	1.98E-01
Kr-87	6.50E+01	1.95E+03	1.95E+02
Kr-88	6.50E+01	1.95E+03	1.95E+02
Kr-89	4.00E+02	1.20E+04	1.20E+03
Xe-131m	4.90E-02	1.47E+00	1.47E-01
Xe-133m	9.80E-01	2.94E+01	2.94E+00
Xe-133	2.70E+01	8.10E+02	8.10E+01
Xe-135m	8.50E+01	2.55E+03	2.55E+02
Xe-135	7.20E+01	2.16E+03	2.16E+02
Xe-137	4.70E+02	1.41E+04	1.41E+03
Xe-138	3.00E+02	9.00E+03	9.00E+02
N-16	4.22E+04	1.27E+06	1.25E+05

Table 12.2-10b

Offgas System

Isotopic Inventory for Preheater Through Charcoal Tanks in MBq

Isotope	Preheater (MBq)	Recombiner (MBq)	Condenser (MBq)	Cooler (MBq)	Tank1 (MBq)	Tank2 (MBq)	Tank3 (MBq)
Kr-83m	7.40E+01	3.10E+01	2.80E+03	7.60E+02	4.30E+05	1.20E+05	1.20E+00
Kr-85m	1.40E+02	5.60E+01	5.10E+03	1.40E+03	1.10E+06	1.30E+06	1.00E+04
Kr-85	5.10E-01	2.10E-01	1.80E+01	5.10E+00	5.00E+03	3.60E+04	3.60E+04
Kr-87	4.10E+02	1.80E+02	1.50E+04	4.20E+03	1.90E+06	2.30E+05	9.70E-03
Kr-88	4.20E+02	1.80E+02	2.60E+02	1.90E+01	3.00E+06	1.70E+06	8.20E+02
Kr-89	2.60E+03	1.00E+03	8.50E+04	2.10E+04	3.10E+05	0	0
Xe-131m	3.10E-01	1.30E-01	1.20E+01	3.30E+00	4.60E+04	2.00E+05	7.10E+04
Xe-133m	6.00E+00	2.50E+00	2.10E+02	6.10E+01	7.00E+05	6.20E+05	2.30E+03
Xe-133	1.80E+02	7.30E+01	6.70E+03	1.80E+03	2.40E+07	6.10E+07	5.70E+06
Xe-135m	5.60E+02	2.30E+02	2.00E+04	5.60E+03	5.00E+05	0	0
Xe-135	4.80E+02	2.00E+02	1.80E+04	5.00E+03	1.70E+07	2.30E+05	0
Xe-137	3.10E+03	1.30E+03	1.10E+05	2.70E+04	4.90E+05	0	0
Xe-138	2.00E+03	8.10E+02	7.10E+04	1.80E+04	1.60E+06	0	0

Table 12.2-11

Turbine Condenser Inventory

Isotope	MBq	Isotope	MBq
Kr-83m	8.82E+03	Na-24	5.32E+01
Kr-85m	1.49E+04	P-32	1.09E+00
Kr-85	5.89E+01	Cr-51	8.16E+01
Kr-87	4.93E+04	Mn-54	9.52E-01
Kr-88	4.93E+04	Mn-56	6.24E+02
Kr-89	3.14E+05	Fe-55	2.73E+01
Xe-131m	4.93E+01	Fe-59	8.16E-01
Xe-133m	7.32E+02	Co-58	2.73E+00
Xe-133	2.09E+04	Co-60	5.44E+00
Xe-135m	6.58E+04	Ni-63	2.73E-02
Xe-135	5.68E+04	Cu-64	7.95E+01
Xe-137	3.89E+05	Zn-65	2.73E+01
Xe-138	2.25E+05	Sr-89	2.73E+00
Total	**1.21E+06**	Sr-90	1.90E-01
		Y-90	1.90E-01
I-131	1.29E+03	Sr-91	1.05E+02
I-132	1.21E+04	Sr-92	2.50E+02
I-133	8.72E+03	Y-91	1.09E+00
I-134	2.23E+04	Y-92	1.52E+02
I-135	1.26E+04	Y-93	1.05E+02
Total	**5.70E+04**	Zr-95	2.18E-01
		Nb-95	2.18E-01
Rb-89	1.14E+02	Mo-99	5.40E+01
Cs-134	7.34E-01	Tc-99m	5.40E+01
Cs-136	4.89E-01	Ru-103	5.44E-01
Cs-137	1.95E+00	Rh-103m	5.44E-01
Cs-138	2.28E+02	Ru-106	8.16E-02
Ba-137m	1.95E+00	Rh-106	8.16E-02
Total	**3.47E+02**	Ag-110m	2.73E-02
		Te-129m	1.09E+00
N-16	3.04E+07	Te-131m	2.69E+00
		Te-132	2.71E-01
H-3	6.08E+04	Ba-140	1.09E+01
		La-140	1.09E+01
		Ce-141	8.16E-01
		Ce-144	8.16E-02
		Pr-144	8.16E-02
		W-187	8.03E+00
		Np-239	2.17E+02
		Total	**1.88E+03**

Table 12.2-12

Isotopic inventory in the ion exchanger filters

Isotope	MBq
Class 2	
I-131	1.64E+06
I-132	1.96E+05
I-133	1.29E+06
I-134	1.39E+05
I-135	6.01E+05
Class 3	
Rb-89	2.08E+02
Cs-134	2.57E+03
Cs-136	8.68E+02
Cs-137	6.94E+03
Cs-138	8.73E+02
Ba-137m	5.92E-01
Class 6	
Sr-89	7.95E+03
Sr-90	6.77E+02
Y-90	1.45E+00
Sr-91	7.24E+03
Sr-92	4.78E+03
Y-91	3.26E+03
Y-92	3.84E+03
Y-93	7.58E+03
Zr-95	6.65E+02
Nb-95	5.86E+02
Mo-99	1.65E+05
Tc-99m	2.31E+03
Ru-103	1.51E+03
Rh-103m	3.62E+00
Ru-106	2.82E+02
Rh-106	4.83E-03
Te-129m	2.90E+03
Te-131m	6.91E+03
Te-132	1.50E+02
Ba-140	1.91E+04
La-140	3.12E+03
Ce-141	2.14E+03
Ce-144	2.80E+02
Pr-144	1.67E-01
Np-239	8.71E+04

Table 12.2-13a

Liquid Waste Management System Equipment Drain Collection Tank

Source Volume = 140 m^3

Total Activity = 6.83E+03 MBq/m^3 = 1.85E-01 µCi/cc

Halogens			Soluble Fission Products			Insoluble Fission Products			Activation Products		
Isotope	**MBq/m^3**	**µCi/cc**	**Isotope**	**MBq/m^3**	**µCi/cc**	**Isotope**	**MBq/m^3**	**µCi/cc**	**Isotope**	**MBq/m^3**	**µCi/cc**
I-131	8.50E+01	2.30E-03	Rb-89	1.52E+02	4.11E-03	Y-91	1.50E+00	4.06E-05	Na-24	7.17E+01	1.94E-03
I-132	8.00E+02	2.16E-02	Sr-89	3.81E+00	1.03E-04	Y-92	2.08E+02	5.61E-03	P-32	1.47E+00	3.98E-05
I-133	6.08E+02	1.64E-02	Sr-90	2.63E-01	7.09E-06	Y-93	1.39E+02	3.75E-03	Cr-51	1.11E+02	2.99E-03
I-134	1.53E+03	4.14E-02	Y-90	2.63E-01	7.09E-06	Zr-95	3.05E-01	8.24E-06	Mn-54	1.27E+00	3.44E-05
I-135	8.57E+02	2.32E-02	Sr-91	1.41E+02	3.80E-03	Nb-95	3.05E-01	8.24E-06	Mn-56	7.92E+02	2.14E-02
			Sr-92	3.39E+02	9.15E-03	Ru-103	7.59E-01	2.05E-05	Fe-55	3.69E+01	9.98E-04
			Mo-99	7.19E+01	1.94E-03	Rh-103M	7.59E-01	2.05E-05	Fe-59	1.12E+00	3.02E-05
			Tc-99M	7.19E+01	1.94E-03	Ru-106	1.12E-01	3.03E-06	Co-58	3.64E+00	9.83E-05
			Te-129M	1.50E+00	4.06E-05	Rh-106	1.12E-01	3.03E-06	Co-60	7.59E+00	2.05E-04
			Te-131M	3.65E+00	9.86E-05	La-140	1.43E+01	3.87E-04	Ni-63	3.69E-02	9.98E-07
			Te-132	3.60E-01	9.74E-06	Ce-141	1.12E+00	3.03E-05	Cu-64	1.07E+02	2.89E-03
			Cs-134	1.01E+00	2.72E-05	Ce-144	1.12E-01	3.03E-06	Zn-65	3.64E+01	9.83E-04
			Cs-136	6.31E-01	1.71E-05	Pr-144	1.12E-01	3.03E-06	Ag-110M	3.69E-02	9.98E-07
			Cs-137	2.59E+00	7.01E-05				W-187	1.11E+01	3.00E-04
			Cs-138	3.12E+02	8.44E-03						
			Ba-140	1.43E+01	3.87E-04						
			Np-239	2.89E+02	7.80E-03						
TOTAL	**3.88E+03**	**1.05E-01**	**TOTAL**	**1.40E+03**	**3.80E-02**	**TOTAL**	**3.66E+02**	**9.89E-03**	**TOTAL**	**1.18E+03**	**3.19E-02**

Table 12.2-13b

Liquid Waste Management System Equipment Drain Sample Tank

Source Volume = 140 m^3

Total Activity = 1.17E+00 MBq/m^3 = 3.17E-05 µCi/cc

Halogens			Soluble Fission Products			Insoluble Fission Products			Activation Products		
Isotope	**MBq/m^3**	**µCi/cc**	**Isotope**	**MBq/m^3**	**µCi/cc**	**Isotope**	**MBq/m^3**	**µCi/cc**	**Isotope**	**MBq/m^3**	**µCi/cc**
I-131	7.82E-02	2.11E-06	Sr-89	3.81E-03	1.03E-07	Y-91	2.17E-03	5.86E-08	Na-24	2.36E-02	6.37E-07
I-132	9.33E-04	2.52E-08	Sr-90	2.63E-04	7.09E-09	Y-91M	2.76E-02	7.45E-07	P-32	1.43E-03	3.85E-08
I-133	2.72E-01	7.34E-06	Y-90	2.63E-04	7.09E-09	Y-92	9.17E-03	2.48E-07	Cr-51	1.08E-01	2.93E-06
I-135	6.62E-02	1.79E-06	Sr-91	2.44E-02	6.59E-07	Y-93	2.78E-02	7.51E-07	Mn-54	1.27E-03	3.44E-08
			Sr-92	7.47E-04	2.02E-08	Zr-95	3.01E-04	8.15E-09	Mn-56	1.27E-03	3.43E-08
			Mo-99	5.39E-02	1.46E-06	Nb-95	3.05E-04	8.24E-09	Fe-55	3.69E-02	9.98E-07
			Tc-99M	5.39E-02	1.46E-06	Ru-103	7.25E-04	1.96E-08	Fe-59	1.09E-03	2.96E-08
			Te-129	2.01E-03	5.44E-08	Rh-103M	7.25E-04	1.96E-08	Co-58	3.64E-03	9.83E-08
			Te-129M	1.47E-03	3.97E-08	Ru-106	1.12E-04	3.03E-09	Co-60	7.59E-03	2.05E-07
			Te-131	8.94E-04	2.42E-08	Rh-106	1.12E-04	3.03E-09	Ni-63	3.69E-05	9.98E-10
			Te-131M	2.11E-03	5.69E-08	La-140	1.43E-02	3.87E-07	Cu-64	2.87E-02	7.77E-07
			Te-132	2.99E-04	8.09E-09	Ce-141	1.12E-03	3.03E-08	Zn-65	3.64E-02	9.83E-07
			Cs-134	1.01E-02	2.72E-07	Ce-144	1.12E-04	3.03E-09	Ag-110M	3.69E-05	9.98E-10
			Cs-136	5.96E-03	1.61E-07	Pr-144	1.12E-04	3.03E-09	W-187	5.42E-03	1.47E-07
			Cs-137	2.59E-02	7.01E-07	Pr-144M	3.65E-06	9.87E-11			
			Ba-136M	1.94E-04	5.24E-09						
			Ba-137M	2.46E-03	6.64E-08						
			Ba-140	1.36E-02	3.68E-07						
			Np-239	2.14E-01	5.78E-06						
TOTAL	**4.17E-01**	**1.13E-05**	**TOTAL**	**4.16E-01**	**1.12E-05**	**TOTAL**	**8.46E-02**	**2.29E-06**	**TOTAL**	**2.56E-01**	**6.91E-06**

Table 12.2-13c

Liquid Waste Management System Floor Drain Collection Tank

Source Volume = 130 m^3

Total Activity = 5.34E+01 MBq/m^3 = 1.44E-03 µCi/cc

Halogens			Soluble Fission Products			Insoluble Fission Products			Activation Products		
Isotope	**MBq/m^3**	**µCi/cc**	**Isotope**	**MBq/m^3**	**µCi/cc**	**Isotope**	**MBq/m^3**	**µCi/cc**	**Isotope**	**MBq/m^3**	**µCi/cc**
I-131	6.79E-01	1.83E-05	Rb-89	1.22E+00	3.29E-05	Y-91	1.19E-02	3.23E-07	Na-24	5.70E-01	1.54E-05
I-132	6.31E+00	1.70E-04	Sr-89	2.92E-02	7.90E-07	Y-92	1.65E+00	4.46E-05	P-32	1.18E-02	3.20E-07
I-133	4.67E+00	1.26E-04	Sr-90	2.04E-03	5.52E-08	Y-93	1.10E+00	2.97E-05	Cr-51	8.75E-01	2.36E-05
I-134	1.20E+01	3.23E-04	Y-90	2.04E-03	5.52E-08	Zr-95	2.34E-03	6.33E-08	Mn-54	1.02E-02	2.76E-07
I-135	6.58E+00	1.78E-04	Sr-91	1.11E+00	3.00E-05	Nb-95	2.34E-03	6.33E-08	Mn-56	6.08E+00	1.64E-04
			Sr-92	2.68E+00	7.23E-05	Ru-103	5.83E-03	1.57E-07	Fe-55	2.92E-01	7.90E-06
			Mo-99	5.65E-01	1.53E-05	Rh-103M	5.83E-03	1.57E-07	Fe-59	8.75E-03	2.36E-07
			Tc-99M	5.65E-01	1.53E-05	Ru-106	9.02E-04	2.44E-08	Co-58	2.92E-02	7.90E-07
			Te-129M	1.19E-02	3.23E-07	Rh-106	9.02E-04	2.44E-08	Co-60	5.83E-02	1.57E-06
			Te-131M	2.89E-02	7.80E-07	La-140	1.14E-01	3.08E-06	Ni-63	2.92E-04	7.90E-09
			Te-132	2.90E-03	7.85E-08	Ce-141	9.02E-03	2.44E-07	Cu-64	8.52E-01	2.30E-05
			Cs-134	7.87E-03	2.13E-07	Ce-144	9.02E-04	2.44E-08	Zn-65	2.92E-01	7.90E-06
			Cs-136	4.98E-03	1.35E-07	Pr-144	9.02E-04	2.44E-08	Ag-110M	2.92E-04	7.90E-09
			Cs-137	2.10E-02	5.66E-07				W-187	8.80E-02	2.38E-06
			Cs-138	2.45E+00	6.61E-05						
			Ba-140	1.14E-01	3.08E-06						
			Np-239	2.32E+00	6.28E-05						
TOTAL	**3.02E+01**	**8.16E-04**	**TOTAL**	**1.11E+01**	**3.01E-04**	**TOTAL**	**2.90E+00**	**7.84E-05**	**TOTAL**	**9.17E+00**	**2.48E-04**

Table 12.2-13d

Liquid Waste Management System Floor Drain Sample Tank

Source Volume = 130 m^3

Total Activity = 9.28E-03 MBq/m^3 = 2.51E-07 µCi/cc

Halogens			Soluble Fission Products			Insoluble Fission Products			Activation Products		
Isotope	**MBq/m^3**	**µCi/cc**	**Isotope**	**MBq/m^3**	**µCi/cc**	**Isotope**	**MBq/m^3**	**µCi/cc**	**Isotope**	**MBq/m^3**	**µCi/cc**
I-131	6.27E-04	1.69E-08	Sr-89	2.92E-05	7.90E-10	Y-91	1.74E-05	4.71E-10	Na-24	1.87E-04	5.05E-09
I-132	7.33E-06	1.98E-10	Sr-90	2.04E-06	5.52E-11	Y-91M	2.17E-04	5.87E-09	P-32	1.11E-05	3.01E-10
I-133	2.20E-03	5.93E-08	Y-90	2.04E-06	5.52E-11	Y-92	7.41E-05	2.00E-09	Cr-51	8.31E-04	2.25E-08
I-135	5.38E-04	1.45E-08	Sr-91	1.90E-04	5.15E-09	Y-93	2.19E-04	5.93E-09	Mn-54	1.02E-05	2.76E-10
			Sr-92	5.73E-06	1.55E-10	Zr-95	2.34E-06	6.33E-11	Mn-56	9.73E-06	2.63E-10
			Mo-99	4.41E-04	1.19E-08	Nb-95	2.34E-06	6.33E-11	Fe-55	2.92E-04	7.90E-09
			Tc-99M	4.14E-04	1.12E-08	Ru-103	5.83E-06	1.57E-10	Fe-59	8.58E-06	2.32E-10
			Te-129	1.57E-05	4.25E-10	Rh-103M	5.83E-06	1.57E-10	Co-58	2.88E-05	7.78E-10
			Te-129M	1.17E-05	3.15E-10	Ru-106	8.48E-07	2.29E-11	Co-60	5.83E-05	1.57E-09
			Te-131	7.44E-06	2.01E-10	Rh-106	8.88E-07	2.40E-11	Ni-63	2.92E-07	7.90E-12
			Te-131M	1.59E-05	4.29E-10	La-140	1.13E-04	3.04E-09	Cu-64	2.31E-04	6.23E-09
			Te-132	2.35E-06	6.36E-11	Ce-141	8.75E-06	2.36E-10	Zn-65	2.88E-04	7.78E-09
			Cs-134	7.87E-05	2.13E-09	Ce-144	8.88E-07	2.40E-11	Ag-110M	2.92E-07	7.90E-12
			Cs-136	4.71E-05	1.27E-09	Pr-144	8.88E-07	2.40E-11	W-187	4.35E-05	1.18E-09
			Cs-137	2.10E-04	5.66E-09	Pr-144M	2.86E-08	7.73E-13			
			Ba-136M	1.52E-06	4.10E-11						
			Ba-137M	1.96E-05	5.31E-10						
			Ba-140	1.09E-04	2.93E-09						
			Np-239	1.64E-03	4.43E-08						
TOTAL	**3.37E-03**	**9.10E-08**	**TOTAL**	**3.24E-03**	**8.76E-08**	**TOTAL**	**6.69E-04**	**1.81E-08**	**TOTAL**	**2.00E-03**	**5.40E-08**

Table 12.2-13e

Liquid Waste Management System Chemical Collection Tank

Source Volume = 4 m^3

Total Activity = 3.24E+02 MBq/m^3 = 8.74E-03 µCi/cc

Halogens			Soluble Fission Products			Insoluble Fission Products			Activation Products		
Isotope	**MBq/m^3**	**µCi/cc**	**Isotope**	**MBq/m^3**	**µCi/cc**	**Isotope**	**MBq/m^3**	**µCi/cc**	**Isotope**	**MBq/m^3**	**µCi/cc**
I-131	4.10E+00	1.11E-04	Rb-89	7.28E+00	1.97E-04	Y-91	7.22E-02	1.95E-06	Na-24	3.35E+00	9.06E-05
I-132	3.82E+01	1.03E-03	Sr-89	1.73E-01	4.67E-06	Y-92	9.97E+00	2.69E-04	P-32	7.05E-02	1.90E-06
I-133	2.83E+01	7.65E-04	Sr-90	1.22E-02	3.30E-07	Y-93	6.61E+00	1.79E-04	Cr-51	5.30E+00	1.43E-04
I-134	7.27E+01	1.96E-03	Y-90	1.22E-02	3.30E-07	Zr-95	1.42E-02	3.84E-07	Mn-54	6.17E-02	1.67E-06
I-135	4.02E+01	1.09E-03	Sr-91	6.69E+00	1.81E-04	Nb-95	1.42E-02	3.84E-07	Mn-56	3.82E+01	1.03E-03
			Sr-92	1.63E+01	4.40E-04	Ru-103	3.50E-02	9.47E-07	Fe-55	1.75E+00	4.72E-05
			Mo-99	3.34E+00	9.03E-05	Rh-103M	3.50E-02	9.47E-07	Fe-59	5.10E-02	1.38E-06
			Tc-99M	3.34E+00	9.03E-05	Ru-106	5.37E-03	1.45E-07	Co-58	1.74E-01	4.70E-06
			Te-129M	7.22E-02	1.95E-06	Rh-106	5.37E-03	1.45E-07	Co-60	3.62E-01	9.78E-06
			Te-131M	1.75E-01	4.72E-06	La-140	6.61E-01	1.79E-05	Ni-63	1.75E-03	4.72E-08
			Te-132	1.74E-02	4.69E-07	Ce-141	5.37E-02	1.45E-06	Cu-64	4.94E+00	1.34E-04
			Cs-134	4.73E-02	1.28E-06	Ce-144	5.37E-03	1.45E-07	Zn-65	1.74E+00	4.70E-05
			Cs-136	2.96E-02	7.99E-07	Pr-144	5.37E-03	1.45E-07	Ag-110M	1.75E-03	4.72E-08
			Cs-137	1.23E-01	3.33E-06				W-187	5.09E-01	1.38E-05
			Cs-138	1.43E+01	3.86E-04						
			Ba-140	6.61E-01	1.79E-05						
			Np-239	1.35E+01	3.64E-04						
TOTAL	**1.84E+02**	**4.96E-03**	**TOTAL**	**6.60E+01**	**1.78E-03**	**TOTAL**	**1.75E+01**	**4.72E-04**	**TOTAL**	**5.65E+01**	**1.53E-03**

Table 12.2-13f

Liquid Waste Management System Detergent Collection Tank

Source Volume = 15 m^3

Total Activity = 3.89E+02 MBq/m^3 = 1.05E-02 µCi/cc

Halogens			Soluble Fission Products			Insoluble Fission Products			Activation Products		
Isotope	**MBq/m^3**	**µCi/cc**	**Isotope**	**MBq/m^3**	**µCi/cc**	**Isotope**	**MBq/m^3**	**µCi/cc**	**Isotope**	**MBq/m^3**	**µCi/cc**
I-131	4.71E+00	1.27E-04	Rb-89	8.79E+00	2.37E-04	Y-91	8.71E-02	2.35E-06	Na-24	4.11E+00	1.11E-04
I-132	4.55E+01	1.23E-03	Sr-89	2.11E-01	5.70E-06	Y-92	1.20E+01	3.25E-04	P-32	8.28E-02	2.24E-06
I-133	3.41E+01	9.23E-04	Sr-90	1.47E-02	3.98E-07	Y-93	7.91E+00	2.14E-04	Cr-51	6.31E+00	1.71E-04
I-134	8.77E+01	2.37E-03	Y-90	1.47E-02	3.98E-07	Zr-95	1.74E-02	4.70E-07	Mn-54	7.53E-02	2.03E-06
I-135	4.80E+01	1.30E-03	Sr-91	8.01E+00	2.17E-04	Nb-95	1.74E-02	4.70E-07	Mn-56	4.61E+01	1.24E-03
			Sr-92	1.93E+01	5.22E-04	Ru-103	4.25E-02	1.15E-06	Fe-55	2.14E+00	5.78E-05
			Mo-99	4.03E+00	1.09E-04	Rh-103M	4.25E-02	1.15E-06	Fe-59	6.31E-02	1.71E-06
			Tc-99M	4.03E+00	1.09E-04	Ru-106	6.31E-03	1.71E-07	Co-58	2.05E-01	5.53E-06
			Te-129M	8.71E-02	2.35E-06	Rh-106	6.31E-03	1.71E-07	Co-60	4.36E-01	1.18E-05
			Te-131M	2.08E-01	5.63E-06	La-140	8.21E-01	2.22E-05	Ni-63	2.14E-03	5.78E-08
			Te-132	2.10E-02	5.66E-07	Ce-141	6.31E-02	1.71E-06	Cu-64	6.15E+00	1.66E-04
			Cs-134	5.77E-02	1.56E-06	Ce-144	6.31E-03	1.71E-07	Zn-65	2.05E+00	5.53E-05
			Cs-136	3.59E-02	9.70E-07	Pr-144	6.31E-03	1.71E-07	Ag-110M	2.14E-03	5.78E-08
			Cs-137	1.46E-01	3.96E-06				W-187	6.48E-01	1.75E-05
			Cs-138	1.77E+01	4.77E-04						
			Ba-140	8.21E-01	2.22E-05						
			Np-239	1.63E+01	4.40E-04						
TOTAL	**2.20E+02**	**5.95E-03**	**TOTAL**	**7.97E+01**	**2.15E-03**	**TOTAL**	**2.11E+01**	**5.69E-04**	**TOTAL**	**6.83E+01**	**1.85E-03**

Table 12.2-13g

Liquid Waste Management System Detergent Sample Tank

Source Volume = 15 m^3

Total Activity = 6.42E+01 MBq/m^3 = 1.74E-03 µCi/cc

Halogens			Soluble Fission Products			Insoluble Fission Products			Activation Products		
Isotope	MBq/m^3	µCi/cc	Isotope	MBq/m^3	µCi/cc	Isotope	MBq/m^3	µCi/cc	Isotope	MBq/m^3	µCi/cc
I-131	4.47E+00	1.21E-04	Sr-89	2.06E-01	5.57E-06	Y-91	1.21E-01	3.27E-06	Na-24	1.35E+00	3.64E-05
I-132	5.29E-02	1.43E-06	Sr-90	1.47E-02	3.98E-07	Y-91m	1.55E+00	4.18E-05	P-32	8.28E-02	2.24E-06
I-133	1.53E+01	4.15E-04	Y-90	1.47E-02	3.98E-07	Y-93	1.58E+00	4.28E-05	Cr-51	6.15E+00	1.66E-04
I-135	3.88E+00	1.05E-04	Sr-91	1.37E+00	3.71E-05	Zr-95	1.69E-02	4.56E-07	Mn-54	7.53E-02	2.03E-06
			Mo-99	3.13E+00	8.47E-05	Nb-95	1.74E-02	4.70E-07	Fe-55	2.14E+00	5.78E-05
			Tc-99M	3.03E+00	8.20E-05	Nb-95M	5.57E-05	1.51E-09	Fe-59	6.31E-02	1.71E-06
			Te-129	1.13E-01	3.07E-06	Ru-103	4.16E-02	1.12E-06	Co-58	2.05E-01	5.53E-06
			Te-129M	8.23E-02	2.22E-06	Rh-103M	4.16E-02	1.12E-06	Co-60	4.36E-01	1.18E-05
			Te-131	5.26E-02	1.42E-06	Ru-106	6.31E-03	1.71E-07	Ni-63	2.14E-03	5.78E-08
			Te-131M	1.20E-01	3.24E-06	Rh-106	6.31E-03	1.71E-07	Cu-64	1.66E+00	4.49E-05
			Te-132	1.70E-02	4.58E-07	La-140	8.21E-01	2.22E-05	Zn-65	2.05E+00	5.53E-05
			Cs-134	5.77E-02	1.56E-06	Ce-141	6.31E-02	1.71E-06	Ag-110	1.03E-05	2.79E-10
			Cs-136	3.40E-02	9.18E-07	Ce-144	6.31E-03	1.71E-07	Ag-110M	2.11E-03	5.70E-08
			Cs-137	1.46E-01	3.96E-06	Pr-144	6.31E-03	1.71E-07	W-187	3.14E-01	8.48E-06
			Ba-136M	1.13E-02	3.07E-07	Pr-144M	2.06E-04	5.57E-09			
			Ba-137M	1.37E-01	3.70E-06						
			Ba-140	7.73E-01	2.09E-05						
			U-235M	2.27E-06	6.13E-11						
			Np-239	1.23E+01	3.33E-04						
			Pu-239	2.37E-06	6.41E-11						
TOTAL	**2.37E+01**	**6.42E-04**	**TOTAL**	**2.16E+01**	**5.85E-04**	**TOTAL**	**4.28E+00**	**1.16E-04**	**TOTAL**	**1.45E+01**	**3.93E-04**

Table 12.2-14a

Solid Waste Management System Phase Separator

Source Volume = 55 m^3

Total Activity = 1.72E+04 MBq/m^3 = 4.65E-01 µCi/cc

Halogens			Soluble Fission Products			Insoluble Fission Products			Activation Products		
Isotope	**MBq/m³**	**µCi/cc**	**Isotope**	**MBq/m³**	**µCi/cc**	**Isotope**	**MBq/m³**	**µCi/cc**	**Isotope**	**MBq/m³**	**µCi/cc**
I-131	4.28E+02	1.16E-02	Sr-89	1.28E+02	3.45E-03	Y-91	1.04E+02	2.81E-03	Na-24	3.62E+00	9.78E-05
I-132	7.23E-01	1.95E-05	Sr-90	4.79E+01	1.30E-03	Y-91M	1.37E+00	3.71E-05	P-32	1.40E+01	3.78E-04
I-133	7.68E+01	2.08E-03	Y-90	4.79E+01	1.30E-03	Y-92	4.15E+00	1.12E-04	Cr-51	2.14E+03	5.77E-02
I-135	1.10E+00	2.97E-05	Sr-91	1.22E+00	3.29E-05	Y-93	9.91E+00	2.68E-04	Mn-54	1.70E+02	4.61E-03
			Sr-92	1.30E-04	3.51E-09	Zr-93	3.20E-03	8.66E-08	Mn-56	5.84E+00	1.58E-04
			Mo-99	8.61E+01	2.33E-03	Zr-95	3.20E+00	8.64E-05	Fe-55	6.23E+03	1.69E-01
			Tc-99M	8.27E+01	2.24E-03	Nb-95	2.31E+01	6.25E-04	Fe-59	3.71E+01	1.00E-03
			Te-129	4.80E+01	1.30E-03	Nb-95M	2.97E-01	8.04E-06	Co-58	1.92E+02	5.19E-03
			Te-129M	3.54E+01	9.57E-04	Ru-103	2.44E+01	6.59E-04	Co-60	1.33E+03	3.59E-02
			Te-131	4.80E-01	1.30E-05	Rh-103M	2.44E+01	6.59E-04	Ni-63	7.01E+00	1.90E-04
			Te-131M	1.10E+00	2.97E-05	Ru-106	1.62E+01	4.37E-04	Cu-64	3.05E+00	8.24E-05
			Te-132	5.75E-01	1.56E-05	Rh-106	1.62E+01	4.37E-04	Zn-65	4.19E+03	1.13E-01
			Cs-134	1.60E+02	4.33E-03	La-140	1.39E+02	3.77E-03	Ag-110	2.08E-02	5.63E-07
			Cs-136	5.38E+00	1.45E-04	Ce-141	3.02E+01	8.15E-04	Ag-110M	4.32E+00	1.17E-04
			Cs-137	4.71E+02	1.27E-02	Ce-144	1.51E+01	4.07E-04	W-187	3.48E+00	9.41E-05
			Ba-136M	1.72E+00	4.64E-05	Pr-144	1.51E+01	4.07E-04			
			Ba-137M	4.40E+02	1.19E-02	Pr-144M	4.80E-01	1.30E-05			
			Ba-140	1.18E+02	3.19E-03						
			U-235M	2.97E-02	8.04E-07						
			Np-239	2.73E+02	7.39E-03						
			Pu-239	2.97E-02	8.04E-07						
TOTAL	**5.07E+02**	**1.37E-02**	**TOTAL**	**1.95E+03**	**5.27E-02**	**TOTAL**	**4.27E+02**	**1.15E-02**	**TOTAL**	**1.43E+04**	**3.87E-01**

Table 12.2-14b

Solid Waste Management System Spent Resin Tank (includes High and Low activity and Condensate)

Source Volume = 210 m^3 Total Activity = 2.54E+06 MBq/m^3 = 6.87E+01 µCi/cc

Halogens			Soluble Fission Products			Insoluble Fission Products			Activation Products		
Isotope	**MBq/m^3**	**µCi/cc**	**Isotope**	**MBq/m^3**	**µCi/cc**	**Isotope**	**MBq/m^3**	**µCi/cc**	**Isotope**	**MBq/m^3**	**µCi/cc**
I-131	6.98E+04	1.89E+00	Rb-89	1.24E+02	3.36E-03	Y-91	1.37E+04	3.71E-01	Na-24	4.38E+03	1.18E-01
I-132	7.36E+03	1.99E-01	Sr-89	1.89E+04	5.12E-01	Y-91M	5.60E+03	1.51E-01	P-32	2.12E+03	5.74E-02
I-133	5.13E+04	1.39E+00	Sr-90	6.67E+03	1.80E-01	Y-92	6.60E+03	1.78E-01	Cr-51	3.12E+05	8.42E+00
I-134	5.05E+03	1.37E-01	Y-90	6.67E+03	1.80E-01	Y-93	5.79E+03	1.57E-01	Mn-54	2.37E+04	6.41E-01
I-135	2.26E+04	6.12E-01	Sr-91	5.44E+03	1.47E-01	Zr-95	1.84E+03	4.97E-02	Mn-56	8.12E+03	2.20E-01
			Sr-92	3.66E+03	9.89E-02	Nb-95	2.94E+03	7.96E-02	Fe-55	8.28E+05	2.24E+01
			Mo-99	1.97E+04	5.31E-01	Nb-95M	3.57E+01	9.64E-04	Fe-59	5.07E+03	1.37E-01
			Tc-99M	1.90E+04	5.15E-01	Ru-103	2.95E+03	7.97E-02	Co-58	2.53E+04	6.82E-01
			Te-129	6.88E+03	1.86E-01	Rh-103M	2.95E+03	7.97E-02	Co-60	1.82E+05	4.91E+00
			Te-129M	5.14E+03	1.39E-01	Ru-106	2.10E+03	5.67E-02	Ni-63	9.46E+02	2.56E-02
			Te-131	1.91E+02	5.16E-03	Rh-106	2.10E+03	5.67E-02	Cu-64	5.36E+03	1.45E-01
			Te-131M	4.50E+02	1.22E-02	La-140	2.09E+04	5.64E-01	Zn-65	5.83E+05	1.57E+01
			Te-132	1.23E+02	3.33E-03	Ce-141	3.78E+03	1.02E-01	Ag-110	1.21E-01	3.27E-06
			Cs-134	2.20E+04	5.94E-01	Ce-144	1.92E+03	5.18E-02	Ag-110M	5.92E+02	1.60E-02
			Cs-136	8.39E+02	2.27E-02	Pr-144	1.92E+03	5.18E-02	W-187	1.10E+03	2.96E-02
			Cs-137	6.41E+04	1.73E+00	Pr-144M	2.55E+00	6.88E-05			
			Cs-138	6.10E+02	1.65E-02						
			Ba-136M	2.67E+02	7.23E-03						
			Ba-137M	6.41E+04	1.73E+00						
			Ba-140	1.85E+04	5.00E-01						
			Np-239	6.70E+04	1.81E+00						
TOTAL	**1.56E+05**	**4.22E+00**	**TOTAL**	**3.30E+05**	**8.93E+00**	**TOTAL**	**7.51E+04**	**2.03E+00**	**TOTAL**	**1.98E+06**	**5.35E+01**

Table 12.2-15

Airborne Sources Calculation

Calculation Bases	
Noble Gas Source at t=30 min	740 MBq/sec (20,000 μCi/sec)
I^{131} Release Rate	3.7 MBq/sec (100 μCi/sec)
Meteorology χ/Q	2.0E-06 s/m^3
Meteorology D/Q	1.8E-09 m^{-2}
Meteorology Boundary	800 m

Table 12.2-16

Annual Airborne Releases for Offsite Dose Evaluations (MBq)

Nuclide	Reactor Building	Turbine Building	Radwaste Building	Mechanical Vacuum Pump	Turbine Seal	Offgas System	Drywell
Kr-83m						3.1E-03	4.3E+01
Kr-85m	6.7E+04	5.6E+05				2.5E+04	1.8E+02
Kr-85						4.2E+06	4.0E+01
Kr-87	4.5E+04	1.4E+06				8.6E-08	1.7E+02
Kr-88	9.0E+04	2.0E+06				1.2E+02	3.7E+02
Kr-89	4.5E+04	1.3E+07	6.5E+05				4.4E+01
Kr-90							1.7E+01
Xe-131m						3.0E+05	2.2E+01
Xe-133m						1.8E+02	1.0E+02
Xe-133	2.5E+06	3.4E+06	4.9E+06	1.9E+07		8.1E+06	6.1E+03
Xe-135m	1.3E+06	9.0E+06	1.2E+07				4.4E+01
Xe-135	2.9E+06	7.4E+06	6.3E+06	7.4E+06		2.6E-23	1.4E+03
Xe-137	4.0E+06	2.2E+07	1.9E+06				6.6E+01
Xe-138	1.8E+05	2.2E+07	4.5E+04				1.4E+02
Xe-139							2.1E+01
I-131	9.2E+02	4.9E+03	3.3E+02	1.8E+03	4.5E+01		6.5E+01
I-132	8.6E+03	4.6E+04	3.1E+03				9.9E+00
I-133	6.2E+03	3.3E+04	2.2E+03		8.3E+01		6.5E+01
I-134	1.6E+04	8.5E+04	5.7E+03				7.0E+00
I-135	9.0E+03	4.8E+04	3.2E+03				3.0E+01
H-3	1.3E+06	1.3E+06					2.6E+05
C-14						3.5E+05	
Na-24							5.4E+00
P-32							1.3E+00
Ar-41						2.9E+04	
Cr-51	2.5E+01	2.1E+01	1.6E+01				1.2E+02
Mn-54	3.2E+01	1.4E+01	9.2E+01				1.6E+00
Mn-56							1.1E+01
Fe-55							4.7E+01

Table 12.2-16

Annual Airborne Releases for Offsite Dose Evaluations (MBq)

Nuclide	Reactor Building	Turbine Building	Radwaste Building	Mechanical Vacuum Pump	Turbine Seal	Offgas System	Drywell
Fe-59	9.0E+00	2.3E+00	6.9E+00				1.3E+00
Co-58	6.9E+00	2.3E+01	4.6E+00				4.4E+00
Co-60	1.2E+02	2.3E+01	1.6E+02				9.4E+00
Ni-63							4.7E-02
Cu-64							7.0E+00
Zn-65	1.2E+02	1.4E+02	6.9E+00				4.6E+01
Rb-89							2.0E-01
Sr-89	1.2E+00	1.4E+02					4.3E+00
Sr-90	2.3E-01	4.6E-01					3.3E-01
Y-90							3.3E-01
Sr-91							6.8E+00
Sr-92							4.7E+00
Y-91							1.7E+00
Y-92							3.7E+00
Y-93							7.4E+00
Zr-95	2.3E+01	9.2E-01	1.8E+01				3.5E-01
Nb-95	2.3E+02	1.4E-01	9.2E-02				3.3E-01
Mo-99	1.5E+03	4.6E+01	6.9E-02				2.4E+01
Tc-99m							2.2E+00
Ru-103	9.7E+01	1.2E+00	2.3E-02				8.3E-01
Rh-103m							8.3E-01
Ru-106							1.4E-01
Rh-106							1.4E-01
Ag-110m	5.5E-02						1.3E-06
Sb-124	1.2E+00	2.3E+00	1.6E+00				
Te-129m							1.6E+00
Te-131m							5.5E-01
Te-132							1.4E-01
Cs-134	1.1E+02	4.6E+00	5.5E+01				1.3E+00
Cs-136	1.2E+01	2.3E+00					5.8E-01
Cs-137	1.4E+02	2.3E+01	9.2E+01				3.4E+00
Cs-138							7.2E-03
Ba-140	5.1E+02	2.3E+02	9.2E-02				1.3E+01
La-140							1.3E+01

Table 12.2-16

Annual Airborne Releases for Offsite Dose Evaluations (MBq)

Nuclide	Reactor Building	Turbine Building	Radwaste Building	Mechanical Vacuum Pump	Turbine Seal	Offgas System	Drywell
Ce-141	2.1E+01	2.3E+02	1.6E-01				1.2E+00
Ce-144							1.4E-01
Pr-144							1.4E-01
W-187							1.3E+00
Np-239							8.2E+01

Table 12.2-17
Comparison of Airborne Concentrations with 10 CFR 20 Concentration

	Airborne Release	Concentration	10 CFR 20
Nuclide	MBq/yr	Bq/m^3	Bq/m^3
Kr-83m	4.3E+01	2.73E-06	2.E+06
Kr-85m	6.5E+05	4.12E-02	4.E+03
Kr-85	4.2E+06	2.66E-01	3.E+04
Kr-87	1.4E+06	8.87E-02	7.E+02
Kr-88	2.1E+06	1.33E-01	3.E+02
Kr-89	1.4E+07	8.87E-01	4.E+01
Kr-90	1.7E+01	1.08E-06	4.E+01
Xe-131m	3.0E+05	1.90E-02	7.E+04
Xe-133m	2.8E+02	1.77E-05	2.E+04
Xe-133	3.8E+07	2.41E+00	2.E+04
Xe-135m	2.2E+07	1.39E+00	1.E+03
Xe-135	2.4E+07	1.52E+00	3.E+03
Xe-137	2.8E+07	1.77E+00	4.E+01
Xe-138	2.3E+07	1.46E+00	7.E+02
Xe-139	2.1E+01	1.33E-06	4.E+01
I-131	8.1E+03	5.13E-04	7.E+00
I-132	5.8E+04	3.68E-03	7.E+02
I-133	4.2E+04	2.66E-03	4.E+01
I-134	1.1E+05	6.97E-03	2.E+03
I-135	6.0E+04	3.80E-03	2.E+02
H-3	2.8E+06	1.77E-01	4.E+03
C-14	3.5E+05	2.22E-02	1.E+02
Na-24	5.4E+00	3.42E-07	3.E+02
P-32	1.3E+00	8.24E-08	2.E+01
Ar-41	2.9E+04	1.84E-03	4.E+02
Cr-51	1.8E+02	1.14E-05	1.E+03
Mn-54	1.4E+02	8.87E-06	4.E+01
Mn-56	1.1E+01	6.97E-07	7.E+02

Table 12.2-17

Comparison of Airborne Concentrations with 10 CFR 20 Concentration

	Airborne Release	Concentration	10 CFR 20
Nuclide	MBq/yr	Bq/m^3	Bq/m^3
Fe-55	4.7E+01	2.98E-06	1.E+02
Fe-59	1.9E+01	1.20E-06	2.E+01
Co-58	3.9E+01	2.47E-06	4.E+01
Co-60	3.1E+02	1.96E-05	2.E+00
Ni-63	4.7E-02	2.98E-09	4.E+01
Cu-64	7.0E+00	4.44E-07	1.E+03
Zn-65	3.1E+02	1.96E-05	1.E+01
Rb-89	2.0E-01	1.27E-08	7.E+03
Sr-89	1.4E+02	8.87E-06	7.E+00
Sr-90	1.0E+00	6.34E-08	2.E-01
Y-90	3.3E-01	2.09E-08	3.E+01
Sr-91	6.8E+00	4.31E-07	2.E+02
Sr-92	4.7E+00	2.98E-07	3.E+02
Y-91	1.7E+00	1.08E-07	7.E+00
Y-92	3.7E+00	2.34E-07	4.E+02
Y-93	7.4E+00	4.69E-07	1.E+02
Zr-95	4.3E+01	2.73E-06	1.E+01
Nb-95	2.3E+02	1.46E-05	7.E+01
Mo-99	1.6E+03	1.01E-04	7.E+01
Tc-99m	2.2E+00	1.39E-07	7.E+03
Ru-103	9.9E+01	6.27E-06	3.E+01
Rh-103m	8.3E-01	5.26E-08	7.E+04
Ru-106	1.4E-01	8.87E-09	7.E-01
Rh-106	1.4E-01	8.87E-09	4.E+01
Ag-110m	5.5E-02	3.49E-09	4.E+00
Sb-124	5.1E+00	3.23E-07	1.E+01
Te-129m	1.6E+00	1.01E-07	1.E+01
Te-131m	5.5E-01	3.49E-08	4.E+01
Te-132	1.4E-01	8.87E-09	3.E+01
Cs-134	1.7E+02	1.08E-05	7.E+00
Cs-136	1.4E+01	8.87E-07	3.E+01
Cs-137	2.6E+02	1.65E-05	7.E+00
Cs-138	7.2E-03	4.56E-10	3.E+03

Table 12.2-17

Comparison of Airborne Concentrations with 10 CFR 20 Concentration

	Airborne Release	Concentration	10 CFR 20
Nuclide	**MBq/yr**	**Bq/m^3**	**Bq/m^3**
Ba-140	7.5E+02	4.75E-05	7.E+01
La-140	1.3E+01	8.24E-07	7.E+01
Ce-141	2.5E+02	1.58E-05	3.E+01
Ce-144	1.4E-01	8.87E-09	7.E-01
Pr-144	1.4E-01	8.87E-09	7.E+00
W-187	1.3E+00	8.24E-08	4.E+02
Np-239	8.2E+01	5.20E-06	1.E+02

Table 12.2-18

ESBWR Annual Average Doses from Airborne Releases

	Annual Dose (mSv/year)							
PATHWAY	**T. BODY**	**GI-TRACT**	**BONE**	**LIVER**	**KIDNEY**	**THYROID**	**LUNG**	**SKIN**
PLUME	7.58E-03	7.58E-03	7.58E-03	7.58E-03	7.58E-03	7.58E-03	7.71E-03	2.04E-02
GROUND	1.93E-04	1.93E-04	1.93E-04	1.93E-04	1.93E-04	1.93E-04	1.93E-04	2.27E-04
VEGET								
ADULT	1.20E-03	1.19E-03	5.45E-03	1.22E-03	1.20E-03	8.27E-03	1.14E-03	1.13E-03
TEEN	1.87E-03	1.86E-03	8.82E-03	1.94E-03	1.90E-03	1.07E-02	1.81E-03	1.80E-03
CHILD	4.36E-03	4.33E-03	2.12E-02	4.53E-03	4.43E-03	2.09E-02	4.30E-03	4.28E-03
MEAT								
ADULT	4.12E-04	4.35E-04	1.98E-03	4.15E-04	4.10E-04	5.62E-04	4.06E-04	4.05E-04
TEEN	3.43E-04	3.56E-04	1.67E-03	3.48E-04	3.44E-04	4.54E-04	3.41E-04	3.40E-04
CHILD	6.39E-04	6.45E-04	3.16E-03	6.45E-04	6.41E-04	8.07E-04	6.37E-04	6.36E-04
COW MILK								
ADULT	4.99E-04	4.70E-04	2.20E-03	5.25E-04	5.02E-04	4.94E-03	4.58E-04	4.53E-04
TEEN	8.79E-04	8.45E-04	4.07E-03	9.48E-04	9.09E-04	7.94E-03	8.35E-04	8.25E-04
CHILD	2.07E-03	2.02E-03	9.98E-03	2.21E-03	2.14E-03	1.62E-02	2.01E-03	2.00E-03
INFANT	4.26E-03	4.22E-03	1.95E-02	4.57E-03	4.39E-03	3.85E-02	4.20E-03	4.17E-03
GOAT MILK								
ADULT	5.81E-04	4.85E-04	2.25E-03	6.26E-04	5.50E-04	5.86E-03	4.89E-04	4.75E-04
TEEN	9.62E-04	8.66E-04	4.16E-03	1.12E-03	9.83E-04	9.39E-03	8.83E-04	8.54E-04
CHILD	2.15E-03	2.06E-03	1.02E-02	2.49E-03	2.27E-03	1.90E-02	2.09E-03	2.05E-03
INFANT	4.36E-03	4.25E-03	1.99E-02	5.10E-03	4.59E-03	4.55E-02	4.31E-03	4.23E-03
INHALE								
ADULT	5.01E-05	6.17E-05	2.01E-05	6.96E-05	8.51E-05	3.82E-03	1.04E-04	3.46E-05
TEEN	5.33E-05	6.63E-05	2.78E-05	8.25E-05	1.04E-04	4.97E-03	1.41E-04	3.49E-05
CHILD	4.99E-05	5.13E-05	3.74E-05	7.62E-05	9.41E-05	6.07E-03	1.19E-04	3.08E-05
INFANT	3.05E-05	2.83E-05	2.78E-05	5.74E-05	5.86E-05	5.53E-03	8.49E-05	1.77E-05
TOTAL	**Annual Dose (mSv/year)**							
ADULT	1.05E-02	1.04E-02	1.97E-02	1.06E-02	1.05E-02	3.12E-02	1.05E-02	2.31E-02
TEEN	1.19E-02	1.18E-02	2.65E-02	1.22E-02	1.20E-02	4.12E-02	1.19E-02	2.45E-02
CHILD	1.70E-02	1.69E-02	5.24E-02	1.77E-02	1.74E-02	7.08E-02	1.71E-02	2.96E-02
INFANT	1.64E-02	1.63E-02	4.72E-02	1.75E-02	1.68E-02	9.73E-02	1.65E-02	2.90E-02

Annual beta air dose = 1.30E+00 millirads
Annual gamma air dose = 1.14E+00 millirads

Table 12.2-19
Average Annual Liquid Releases

	Annual Release	Concentration	10CFR20 MPC
Nuclide	MBq/yr	Bq/ml	Bq/ml
I-131	8.90E+01	2.98E-05	1.10E-02
I-132	9.30E+01	3.11E-05	3.00E-01
I-133	2.88E+02	9.66E-05	3.70E-02
I-134	7.25E+01	2.43E-05	7.40E-01
I-135	2.38E+02	7.96E-05	1.50E-01
H-3	2.20E+06	7.37E-01	1.10E+02
C-14	5.90E+00	1.98E-06	3.00E+01
Na-24	1.12E+02	3.74E-05	7.40E+00
P-32	6.72E+00	2.25E-06	7.40E-03
Cr-51	6.96E+02	2.33E-04	7.40E+01
Mn-54	2.42E+02	8.11E-05	3.70E+00
Mn-56	4.84E+02	1.62E-04	3.70E+00
Co-56	3.09E+02	1.03E-04	1.10E+02
Co-57	4.39E+00	1.47E-06	3.70E-03
Co-58	8.18E+00	2.74E-06	3.30E+00
Co-60	8.66E+02	2.90E-04	1.10E+00
Fe-55	1.06E+03	3.54E-04	7.40E+01
Fe-59	1.84E+01	6.16E-06	1.90E+00
Ni-63	2.62E+01	8.76E-06	1.10E+00
Cu-64	1.59E+02	5.34E-05	1.10E+01
Zn-65	8.18E+01	2.74E-05	3.70E+00
Rb-89	2.31E+00	7.72E-07	1.10E-01
Sr-89	3.09E+00	1.04E-06	1.10E-01
Sr-90	1.01E+00	3.37E-07	1.10E-02
Y-90	8.51E-02	2.85E-08	7.40E-01
Sr-91	2.97E+01	9.96E-06	2.60E+00
Y-91	3.12E+00	1.05E-06	1.10E+00
Sr-92	3.26E+01	1.09E-05	2.60E+00
Y-92	2.32E+01	7.75E-06	2.20E+00
Y-93	2.94E+01	9.84E-06	1.10E+00
Zr-95	2.43E+01	8.12E-06	2.20E+00
Nb-95	2.89E+01	9.69E-06	3.70E+00
Mo-99	2.43E+01	8.13E-06	7.40E+00
Tc-99m	2.35E+01	7.87E-06	2.20E+02

Table 12.2-19

Average Annual Liquid Releases

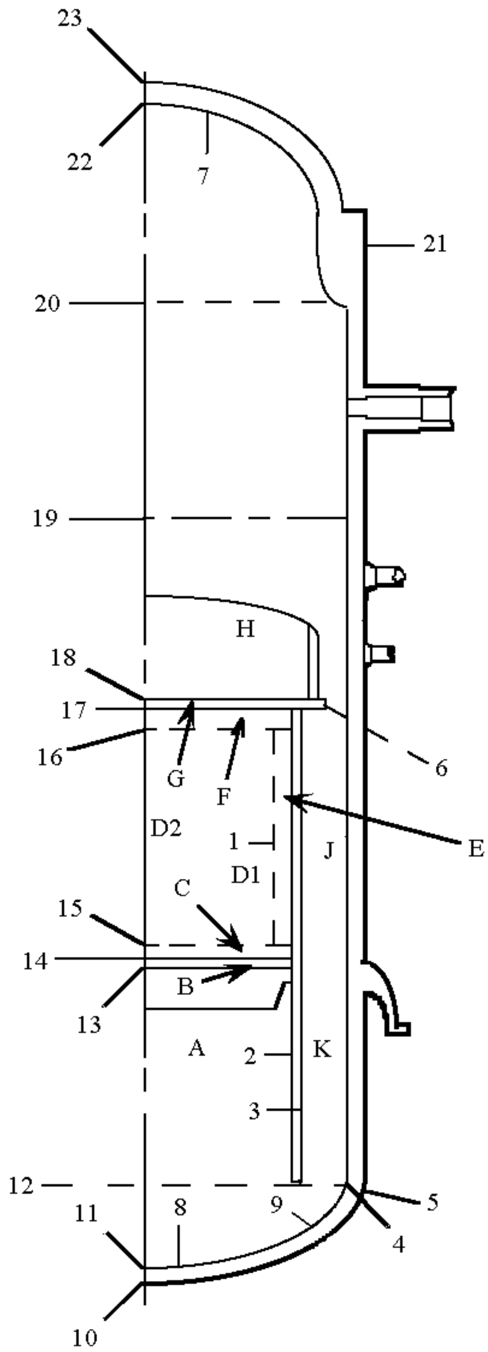
	Annual Release	Concentration	10CFR20 MPC
Nuclide	**MBq/yr**	**Bq/ml**	**Bq/ml**
Ru-103	5.04E+00	1.69E-06	3.00E+00
Rh-103m	2.48E-01	8.31E-08	3.70E+02
Ru-106	3.67E-02	1.23E-08	3.70E-01
Rh-106	4.82E+00	1.61E-06	1.10E-01
Ag-110m	6.04E+01	2.02E-05	1.10E+00
Sb-124	2.11E+01	7.07E-06	7.40E-01
Te-129m	4.80E-01	1.61E-07	1.10E+00
Te-131m	1.07E+00	3.57E-07	2.20E+00
Te-132	1.18E-01	3.95E-08	1.10E+00
Cs-134	1.74E+02	5.83E-05	3.30E-01
Cs-136	9.17E+00	3.07E-06	3.30E+00
Cs-137	2.45E+02	8.22E-05	7.40E-01
Cs-138	9.73E+00	3.26E-06	1.10E-01
Ba-140	1.90E+01	6.37E-06	1.10E+00
La-140	4.79E+00	1.61E-06	7.40E-01
Ce-141	3.36E+00	1.13E-06	3.30E+00
Ce-144	5.35E+01	1.79E-05	3.70E-01
Pr-143	7.80E-02	2.61E-08	1.90E+00
W-187	1.88E+01	6.30E-06	2.20E+00
Np-239	8.54E+01	2.86E-05	3.70E+00
TOTAL	**2.21E+06**		

Table 12.2-20

Liquid Pathway Dose Analysis in mSv/year

	Annual Doses (mSv/yr)							
PATHWAY	**SKIN**	**BONE**	**LIVER**	**T. BODY**	**THYROID**	**KIDNEY**	**LUNG**	**GI-LLI**
Drinking								
Adult		3.25E-04	1.27E-03	1.17E-03	2.24E-03	1.01E-03	9.18E-04	1.29E-03
Teenager		3.06E-04	9.90E-04	7.86E-04	1.81E-03	7.40E-04	6.65E-04	8.93E-04
Child		8.70E-04	1.94E-03	1.40E-03	4.17E-03	1.43E-03	1.27E-03	1.42E-03
Infant		8.47E-04	2.09E-03	1.33E-03	5.86E-03	1.41E-03	1.26E-03	1.31E-03
Fish								
Adult		3.54E-02	2.22E-02	1.59E-02	6.18E-04	7.04E-03	2.25E-03	7.47E-03
Teenager		3.82E-02	2.28E-02	9.66E-03	5.83E-04	7.16E-03	2.66E-03	5.57E-03
Child		4.86E-02	2.01E-02	4.83E-03	6.27E-04	6.03E-03	2.10E-03	2.15E-03
Shoreline								
Adult	8.37E-05	7.13E-05	7.13E-05	7.13E-05	7.13E-05	7.13E-05	7.13E-05	7.13E-05
Teenager	4.67E-04	3.98E-04	3.98E-04	3.98E-04	3.98E-04	3.98E-04	3.98E-04	3.98E-04
Child	9.77E-05	8.31E-05	8.31E-05	8.31E-05	8.31E-05	8.31E-05	8.31E-05	8.31E-05
Total								
Adult	8.37E-05	3.58E-02	2.35E-02	1.71E-02	2.93E-03	8.12E-03	3.24E-03	8.83E-03
Teenager	4.67E-04	3.89E-02	2.42E-02	1.08E-02	2.79E-03	8.30E-03	3.72E-03	6.86E-03
Child	9.77E-05	4.96E-02	2.21E-02	6.31E-03	4.88E-03	7.54E-03	3.45E-03	3.65E-03
Infant		8.47E-04	2.09E-03	1.33E-03	5.86E-03	1.41E-03	1.26E-03	1.31E-03

Figure 12.2-1. Radiation Source Model



Note: See Table 12.2-1 for component designations.

12.3 RADIATION PROTECTION

12.3.1 Facility Design Features

The ESBWR Standard Plant is designed in accordance with Regulatory Guide 8.8, i.e., to keep radiation exposures to plant personnel as low as reasonably achievable (ALARA). This section describes the component and system designs in addition to the equipment layout employed to maintain radiation exposures ALARA. Consideration of individual systems is provided to illustrate the application of these principles. Owing to the ESBWR being a standard plant, specific details as to precise equipment definition are not available and are to be provided by the COL applicant during the final design detail stage. To insure that the plant as designed meets all applicable radiation criteria, a two-step process is then applied where design details not included in this document are then subject to review and confirmation in accordance with radiation protection criteria. Therefore, the details in this section serve as input to the final design configuration and serve to determine the adequacy of the design with respect to radiation protection .

Material selection for primary coolant piping, tubing, vessel internal surfaces, and other components in contact with the primary coolant is discussed in the following pages.

Carbon steel is used in a large portion of the system piping and equipment outside of the nuclear steam supply system. Carbon steel is typically low in nickel content and contains a very small amount of cobalt impurity.

Stainless steel is used in portions of the system such as the reactor internal components and heat exchanger tubes where high corrosion resistance is required. The nickel content of the stainless steels is in the 9 to 10.5% range and is controlled in accordance with applicable ASME material specifications. Cobalt content is controlled to less than 0.05% in the XM-19 alloy used in the control rod drives.

A previous review of materials certifications indicated an average cobalt content of only 0.15% in austenitic stainless steels.

Ni-Cr-Fe alloys such as Inconel 600 and Inconel X750, which have high nickel content, are used in some reactor vessel internal components. These materials are used in applications for which there are special requirements to be satisfied (such as possessing specific thermal expansion characteristics along with adequate corrosion resistance) and for which no suitable alternative low-nickel material is available. Cobalt content in the Inconel X750 used in the fuel assemblies is limited to 0.05%.

Stellite is used for hard facing of components that must be extremely wear resistant. Use of high cobalt alloys such as Stellite is restricted to those applications where no satisfactory alternative material is available. An alternative material (Colmonoy) has been used for some hard facings in the core area.

Main condenser tubes and tubes sheets are made of titanium alloys to minimize the introduction of foreign material into the reactor system as a result of condenser tube leakage .

The COL applicant shall address material selection of systems and components exposed to reactor coolant to maintain radiation exposures ALARA. See Subsection 12.3.7 for COL license information requirements.

12.3.1.1 Equipment Design for Maintaining Exposure ALARA

This subsection describes specific components as well as system design features that aid in maintaining the exposure of plant personnel during system operation and maintenance ALARA. Equipment layouts to provide ALARA exposures of plant personnel are discussed in Subsection 12.3.1.2.

12.3.1.1.1 Pumps

Pumps located in radiation areas are designed to minimize the time required for maintenance. Quick-change cartridge-type seals on pumps, and pumps with back pullout features that permit removal of the pump impeller or mechanical seals without disassembly of attached piping are employed to minimize exposure time during pump maintenance. The configuration of piping about pumps is designed to provide sufficient space for efficient pump maintenance. Toward this end, systems that contain pumps generally have the pumps in a separate alcove with piping routed to the back of the alcove into shield pipe chases. Provisions are made for flushing and in certain cases chemically cleaning pumps prior to maintenance. Pump casing drains provide a means for draining pumps to the sumps prior to disassembly, thus reducing the exposure of personnel and decreasing the potential for contamination. Where two or more pumps conveying highly radioactive fluids are required for operational reasons to be located adjacent to each other, shielding is provided between the pumps to maintain exposure levels ALARA. Pumps adjacent to other highly radioactive equipment are also shielded to reduce the maintenance exposure, for example, in the radwaste system.

Whenever possible, operation of the pumps and associated valves for radioactive systems is accomplished remotely through reach rods or electric controls. Pump control instrumentation is located outside high radiation areas in separate alcoves, and motor- or pneumatic-operated valves and valve extension stems are employed to allow operation from these areas.

12.3.1.1.2 Instrumentation

Instruments are located in low radiation areas such as shielded valve galleries, corridors, or control rooms, whenever possible. Shielded valve galleries provided for this purpose include those for the RWCU/SDC, FAPCS, and radwaste (cleanup phase separator and spent resin tank) systems. Instruments that are required to be located in high radiation areas due to operations requirements are designed such that removal of these instruments to low radiation areas for maintenance is possible. Sensing lines are routed from taps on the primary system in order to avoid placing the transmitters or readout devices in high radiation areas. For example, reactor water level sensing instruments are located outside the drywell.

Liquid service equipment for systems containing radioactive fluids is provided with vent and backflush provisions. Instrument lines, except those for the reactor vessel, are designed with provisions for backflushing and maintaining a clean fill in the sensing lines. The reactor vessel sensing lines may be flushed with condensate following reactor blowdown.

12.3.1.1.3 Heat Exchangers

Heat exchangers are constructed of stainless steel or Cu/Ni tubes to minimize the possibility of failure and reduce maintenance requirements. The heat exchanger design allows for the complete drainage of fluids from the exchanger, avoiding pooling effects that could lead to radioactive crud deposition. Connections are available for condensate or demineralized water flushing of the heat exchangers. For the RWCU/SDC, separate connections are also provided for introducing chemical cleaning solutions for decontaminating the heat exchangers. The fuel pool heat exchanger is downstream of the filter/demineralizer and is therefore not subjected to flows containing significant amounts of fission or activation products.

Instrumentation and valves are remotely operable to the maximum extent possible in the shielded heat exchanger cubicles, to reduce the need for entering these high radiation areas.

12.3.1.1.4 Valves

Valve packing and gasket material are selected on a conservative basis, accounting for environmental conditions such as temperature, pressure, and radiation tolerance requirements to provide a long operating life. Valves have back seats to minimize the leakage through the packing. Straight-through valve configurations were selected where practical, over those that exhibit flow discontinuities or internal crevices to minimize crud trapping. Teflon gaskets are not used.

Wherever possible, valves in systems containing radioactive fluids are separated from those for “clean” services to reduce the radiation exposure from adjacent valves and piping during maintenance.

Pneumatic or mechanically operated valves are employed in high radiation areas, whenever practical, to minimize the need for entering these areas. For certain situations, manually operated valves are required, and in such cases extension valve stems are provided which are operated from a shielded area. Flushing and drain provisions are employed in radioactive systems to reduce exposure to personnel during maintenance.

For areas in which especially high radiation levels are encountered, valves are reduced to the maximum extent possible with the bulk of the valve and piping located in an adjacent valve gallery where the radiation levels are lower.

12.3.1.1.5 Piping

Piping was selected to provide a service life equivalent to the design life of the plant, with consideration given to corrosion allowances and environmental conditions. Piping in radioactive systems such as the RWCU/SDC has butt-welded connections, rather than socket welds, to reduce crud traps. Distinction is made between piping conveying radioactive and non radioactive fluids, and separate routing through shielded pipe chases is provided whenever possible. Piping conveying highly contaminated fluids is usually routed through shielded pipe chases and shielded cubicles. However, when these options are not feasible, the radioactive piping is embedded in concrete walls and floors.

12.3.1.1.6 Lighting

Lighting is designed to provide sufficient illumination in radiation areas to allow quick and efficient surveillance and maintenance operations. To reduce the need for immediate replacement of defective bulbs, multiple lighting fixtures are provided in shielded cubicles. Incandescent lamps are the only type of lamp used within the primary containment, the main steam tunnel, and the refueling level of the reactor building. They require less time for servicing and, hence, the personnel exposure is reduced. Consideration is also given to locating lighting fixtures in easily accessible locations, thus reducing the exposure time for bulb replacement.

12.3.1.1.7 Floor Drains

Floor drains with appropriately sloped floors are provided in shielded cubicles where the potential for spills exist. Those drain lines having a potential for containing highly radioactive fluids are routed through pipe chases, shielded cubicles, or are embedded in concrete walls and floors. Smooth epoxy-type coatings are employed to facilitate decontamination when a spill does occur.

12.3.1.1.8 Ventilation

The Controlled Area Ventilation Subsystem of Reactor Building HVAC (CONAVS) supplies air to the containment during reactor shutdown for personnel access to the containment area. During normal operation, the preheated outside air travels through the Air Handling Units (AHU) where particulates are removed from the air by low and high efficiency filters; heat is transferred between the air and the hot/chilled water coils; and the conditioned air is distributed to the controlled areas by the supply fan.

The exhaust subsystem consists of redundant exhaust fans connected to common collection and discharge duct systems. During normal operation, the operating fan exhausts air from the controlled areas directly to the atmosphere through the plant vent stack. During purge operation, the operating purge fan exhausts air from the containment area through the purge exhaust filter unit prior to discharge to the plant vent stack.

The Refueling and Pool Area Ventilation Subsystem of Fuel Building HVAC (REPAVS) is a once-through ventilation system that distributes conditioned air to the refueling area of the reactor and spent fuel pool area of the Fuel Building. During normal operation, outside air travels through the AHU's stages where particulates are removed from the air by low and high efficiency filters; heat is transferred between the air and the hot/chilled water coils; and the conditioned air is distributed to the refueling area and spent fuel pool surfaces. Air is ducted to the exhaust fan and exhausted to the outside atmosphere through the plant vent stack. The exhaust system has the manual capability to divert the exhaust for filtration by the purge exhaust filter unit, prior to discharge to the plant vent stack.

12.3.1.2 Plant Design for Maintaining Exposure ALARA

This subsection describes features of equipment layout and design that are employed to maintain personnel exposures ALARA.

12.3.1.2.1 Penetrations

Penetrations through shield walls are avoided whenever possible to reduce the number of streaming paths provided by these penetrations. Whenever penetrations are required through shield walls, however, they are located to minimize the effect on surrounding areas. Penetrations are located so that the radiation source cannot "see" through the penetration. When this is not possible, or to provide an added order of reduction, penetrations are located to exit far above floor level in open corridors or in other relatively inaccessible areas. Penetrations that are offset through a shield wall are frequently employed for electrical penetrations to reduce the streaming of radiation through these penetrations.

Where permitted, the annular region between pipe and penetration sleeves, as well as electrical penetrations, are filled with shielding material to reduce the streaming area presented by these penetrations. The shielding materials used in these applications include lead-loaded silicone foam, with a density comparable to concrete, and boron-loaded refractory-type material for applications requiring neutron as well as gamma shielding. There are certain penetrations where these two approaches are not feasible or are not sufficiently effective. In those cases, a shielded enclosure about the penetration as it exits in the shield wall, with a 90-degree bend of the process pipe as it exits the penetration, is employed.

12.3.1.2.2 Sample Stations

Sample stations in the plant provide for the routine surveillance of reactor water quality. These sample stations are located in low radiation areas to reduce the exposure to operating personnel. Flushing provisions are included using demineralized water, and pipe drains to plant sumps are provided to minimize the possibility of spills. Fume hoods are employed for airborne contamination control. Both working areas and fume hoods are constructed of polished stainless steel to ease decontamination if a spill does occur. Grab spouts are located above the sink to reduce the possibility of contaminating surrounding areas during the sampling process.

12.3.1.2.3 HVAC Systems

Major HVAC equipment (blowers, coolers, and the like) is located in dedicated low radiation areas to minimize exposures to personnel maintaining this equipment ALARA. HVAC ducting is routed outside pipe chases and does not penetrate pipe chase walls, which could compromise the shielding. HVAC ducting penetrations through walls of shielded cubicles are located to minimize the effect of the streaming radiation levels in adjoining areas. Additional HVAC design considerations are addressed in Subsection 12.3.3.

12.3.1.2.4 Piping

Piping containing radioactive fluids is routed through shielded pipe chases, shielded equipment cubicles, or embedded in concrete walls and floors, whenever possible. Where possible, “clean” services, such as compressed air and demineralized water, are not routed through shielded pipe chases. For situations in which radioactive piping must be routed through corridors or other low radiation areas, an analysis is conducted to ensure this routing does not compromise the existing radiation zoning.

Radioactive services are routed separately from piping containing nonradioactive fluids, whenever possible, to minimize the exposure to personnel during maintenance. When such routing combinations are required, however, drain provisions are provided to remove the radioactive fluid contained in equipment and piping. In such situations, provisions are made for the valves required for process operation to be controlled remotely, without need for entering the cubicle.

“Clean” services and radioactive piping are required at times to be routed together in shielded cubicles. In such situations, provisions are made for the valves required for process operation to be controlled remotely, without need for entering the cubicle.

Penetrations for piping through shield walls are designed to minimize the effect on surrounding areas. Approaches used to accomplish this objective are described in Subsection 12.3.1.2.

Piping configurations are designed to minimize the number of "dead legs" and low points in piping runs to avoid accumulation of radioactive crud and fluids in the line. Drains and flushing provisions are employed whenever feasible to reduce the effect of required "dead legs" and low points. Systems containing radioactive fluids are welded to the most practical extent to reduce leakage through flanged or screwed connections. For highly radioactive systems, butt welds are employed to minimize crud traps. Provisions are also made in radioactive systems for flushing with condensate or chemically cleaning the piping to reduce crud buildup.

12.3.1.2.5 Equipment Layout

Equipment layout is designed to reduce the exposure of personnel required to inspect or maintain equipment. "Clean" pieces of equipment are located separately from those which are sources of radiation whenever possible. For systems that have components that are major sources of radiation, piping and pumps are located in separate cubicles to reduce exposure from these components during maintenance. These major radiation sources are also separately shielded from each other.

12.3.1.2.6 Contamination Control

Contaminated piping systems are welded to the most practical extent to minimize leaks through screwed or flanged fittings. For systems containing highly radioactive fluids, drains are hard piped directly to equipment drain sumps, rather than to allow contaminated fluid to flow across the floor to a floor drain. Certain valves in the main steam line are also provided with leakage drains piped to equipment drain sumps to reduce contamination of the steam tunnel. Pump casing drains are employed on radioactive systems whenever possible to remove fluids from the pump prior to disassembly. In addition, provisions for flushing with condensate, and in especially contaminated systems, for chemically cleaning the equipment prior to maintenance, are provided.

The HVAC System is designed to limit the extent of airborne contamination by providing air-flow patterns from areas of low contamination to more contaminated areas. This, in general, is accomplished by pressurizing the main corridor on each floor with the flow directed outward in each cubicle and then to the pipe chases where the flow is directed to the plant stack. Penetrations through outer walls of the building containing radiation sources are sealed to prevent miscellaneous leaks into the environment. The equipment drain sump vents are piped directly to the radwaste HVAC System to remove airborne contaminants evolved from discharges to the sump. Wet transfer of both the steam dryer and separator also reduces the likelihood of contaminants on this equipment being released into the plant atmosphere. In areas where the reduction of airborne contaminants cannot be eliminated efficiently by HVAC Systems, breathing air provisions are provided, for example, for CRD removal under the reactor pressure vessel and in the CRD maintenance room.

Appropriately sloped floor drains are provided in shielded cubicles and other areas where the potential for a spill exists to limit the extent of contamination. Curbs are also provided to limit contamination and simplify washdown operations. A cask decontamination vault is located in the reactor building where the spent fuel cask and other equipment may be cleaned. The CRD maintenance room is used for disassembling control rod drives to reduce the contamination potential.

Consideration is given in the design of the plant for reducing the effort required for decontamination. Epoxy-type wall and floor coverings have been selected which provide smooth surfaces to ease decontamination. Expanded metal-type floor gratings are minimized in favor of smooth surfaces in areas where radioactive spills could occur. Equipment and floor drain sumps are stainless steel lined to reduce crud buildup and to provide surfaces easily decontaminated.

12.3.1.3 Radiation Zoning

Radiation zones are established in all areas of the plant as a function of both the access requirements of that area and the radiation sources in that area. Operating activities, inspection requirements of equipment, maintenance activities, and abnormal operating conditions are considered in determining the appropriate zoning for a given area. The relationship between radiation zone designations and accessibility requirements is presented in the following tabulation:

Zone Designation	Dose Rate µSv/hr	Access Description
A	< 6	Uncontrolled, unlimited access
B	< 10	Controlled, unlimited access
C	< 50	Controlled, limited access, 20 hr/wk.
D	< 250	Controlled, limited access, 4 hr/wk.
E	< 1000	Controlled, limited access, 1 hr/wk.
F or higher	> 1000	Controlled access. Authorization required.

The dose rate applicable for a particular zone is based on operating experience and represents design dose rates in a particular zone, and should not be interpreted as the expected dose rates which would apply in all portions of that zone, or for all types of work within that zone, or at all periods of entry into the zone. Large BWR plants have been in operation for three decades, and operating experience with similar design basis numbers shows that only a small fraction of the 10 CFR 20 maximum permissible dose is received in such zones from radiation sources controlled by equipment layout or the structural shielding provided. Therefore, on a practical basis, a radiation zoning approach as described above accomplishes the as low as reasonably achievable objectives for doses as required by 10 CFR 20 Subpart C. The radiation zone maps for this plant with zone designations as described in the preceding tabulations are contained in Figures 12.3-1 through 12.3-22

Access to areas in the plant is controlled and regulated by the zoning of a given area. Areas with dose rates such that an individual would receive a dose in excess of 1000 µSv (100 mRem) in a period of one hour are locked and posted with "High Radiation Area" signs. Entry to these areas is on a controlled basis. Areas in which an individual would receive a dose in excess of 50 µSv (5 mRem) up to 1000 µSv (100 mRem) within a period of one hour are posted with signs indicating that this is a radiation area and include, in certain cases, barriers such as ropes or doors.

12.3.1.4 Implementation of ALARA

In this subsection, the implementation of design considerations to radioactive systems for maintaining personnel radiation exposures as low as reasonably achievable is described for the RWCU/SDC, Main Steam, and FAPCS.

12.3.1.4.1 Reactor Water Cleanup / Shutdown Cooling System

This system is designed to operate continuously to reduce reactor water radioactive contamination, as well as perform shutdown cooling. Components for this system are located outside the containment and include demineralizers, regenerative (RHX) and nonregenerative (NRHX) heat exchangers, pumps, and associated valves.

The highest radiation level components include the demineralizers and heat exchangers. The demineralizers are located in separate concrete-shielded cubicles that are accessible through shielded hatches. Valves and piping within the cubicles are reduced to the extent that entry into the cubicles is not required during any operational phase. Most of the valves and piping are located in a shielded valve gallery adjacent to the demineralizer cubicles. The valves are remotely operable to the greatest practical extent to minimize entry requirements into this area. The RWCU/SDC heat exchangers are also located in a shielded cubicle with valves operated remotely by use of extension valve stems, or from instrument panels located outside the cubicle. The backwash tank is shielded separately from the resin transfer pump, permitting maintenance of the pump without being exposed to the spent resins contained in the backwash tank. The pump valves are operated remotely from outside the cubicle. The resin pump valves are operated remotely from outside the cubicle.

The RWCU/SDC is provided with chemical cleaning and decontamination connections that can utilize the condensate system to flush piping and equipment prior to maintenance to provide decontamination of pumps, the shell side of the RHX, the tube side of both the RHX and NRHX taken together. The RWCU/SDC demineralizer can be remotely back-flushed to remove spent resins. If additional decontamination is required, chemical addition connections are provided in the piping to clean piping as well as equipment prior to maintenance. The HVAC System is designed to limit the spread of contaminants from these shielded cubicles by maintaining a negative pressure in the cubicles relative to the surrounding areas.

Personnel access to the cubicles for maintenance of these components is on a controlled basis whereby specific restrictions and controls are implemented to minimize personnel exposure.

12.3.1.4.2 Fuel and Auxiliary Pools Cooling System

This system is designed to operate continuously to handle the spent fuel cooling load and to reduce pool water radioactive contamination in all the major pools in the ESBWR. The system components are located in the fuel and reactor building. Included are two independent filter demineralizer units that serve to remove radioactive contamination from the fuel pool and suppression water during cleanup and Low Pressure Coolant Injection (LPCI) mode. These units are the highest radiation level components in the system. Each unit is located in a concrete-shielded cubicle that is accessible through a shielded hatch. Provisions are made for remotely backflushing the units when filter and resin material are spent. This removal of radioactivity contaminated material reduces the component radiation level considerably and serves to minimize exposures during maintenance. All valves (inlet, outlet, recycle, vent, and drain) to the filter demineralizer units are located outside the shielded cubicles in a separate shielded cubicle together with associated piping, headers, and instrumentation. The radiation level in this cubicle is sufficiently low to permit required maintenance to be performed. Piping potentially containing resin is continuously sloped downward to the backwash tank. The system also includes two low radiation level heat exchangers and two circulation pumps.

All of the aforementioned shielded system components are consolidated in the same section of the reactor building. Personnel access to shielded system components is controlled to minimize personnel exposure. Shielding for the components is designed to reduce the radiation level to less than 1 mR/hr in adjacent areas where normal access is permitted.

Operation of the system is accomplished from the MCR and local control panels located where designed radiation levels are less than 25μSv/hr (2.5 mR/h) and normal personnel access is permitted.

12.3.1.4.3 Main Steam System

All radioactive materials in the main steam system, located in the main steam-feedwater pipe tunnel of the reactor building, result from radioactive sources carried over from the reactor during plant operation, including high energy short-lived Nitrogen-16. During plant shutdown, residual radioactivity from prior plant operation is the radiation source.

Access to the main steam pipe tunnel in the reactor building is controlled. Entry into the reactor building steam tunnel is through a controlled personnel access door shielded by a concrete labyrinth to attenuate radiation streaming from the steam lines to adjoining areas. During reactor operation, the steam tunnel is not accessible except in the hot standby conditions under regulated access.

Providing valve drains that are piped to equipment drain sumps minimizes leakage from selected valves into surrounding areas. Floor drains are provided to minimize the spread of contamination should a leakage occur.

Penetrations through the steam tunnel walls are minimized to reduce the streaming paths made available by these penetrations. Penetrations through the steam tunnel walls, when they are required, are located so as to exit in controlled access areas or in areas that are not aligned with the steam lines. A lead-loaded silicone foam is employed whenever possible for these penetrations to reduce the available streaming area presented.

12.3.2 Shielding

12.3.2.1 General Design Guides

The primary objective of the radiation shielding is to protect operating personnel and the general public from radiation emanating from the reactor, the power conversion systems, the radwaste process systems, and the auxiliary systems, while maintaining appropriate access for operation and maintenance. The radiation shielding is also designed to keep radiation doses to equipment below levels at which disabling radiation damage occurs.

Specifically, the shielding requirements in the plant are designed to perform the following functions:

- limit the exposure of the general public, plant personnel, contractors, and visitors to levels that are ALARA and within 10 CFR 20 requirements;
- limit the radiation exposure of personnel, in the unlikely event of an accident, to levels that are ALARA and which conform to the limits specified in 10 CFR 50, Appendix A, Criterion 19 to ensure that the plant is maintained in a safe condition during an accident; and
- limit the radiation exposure of critical components within specified radiation tolerances, to assure that component performance and design life are not impaired.

12.3.2.2 Design Description

12.3.2.2.1 General Design Guides

In order to meet the above design objectives, the following design guides are used in the shielding design of the ESBWR:

- All systems containing radioactivity are identified and shielded based on access and exposure level requirements of surrounding areas. The radiation zone maps described in Subsection 12.3.1.3 indicate design radiation levels for which shielding for equipment contributing to the dose rate in the area is designed.
- The source terms used in the shielding calculations are analyzed with a conservative approach. Transient conditions as well as shut down and normal operating conditions are considered to ensure that a conservative source is used in the analysis. Shielding design is based on fission product quantities in the coolant corresponding to the design basis off-gas release, in addition to activation products. This is considered an anticipated operational occurrence, and hence represents conservatism in design. For components where N^{16} is the major radiation source, a concentration based upon operating plant data is used.
- Effort is made to locate processing equipment in a manner that minimizes the shielding requirements. Shielded labyrinths are used to eliminate radiation streaming through access ways from sources located in cubicles.
- Penetrations through shield walls are located so as to minimize the effect on surrounding areas due to radiation streaming through the penetrations. The approaches used to locate and shield penetrations, when required, are discussed in Subsection 12.3.1.2.
- Wherever possible, radioactive piping is run in a manner that minimizes radiation exposure to plant personnel. This involves:
  - minimizing radioactive pipe routing in corridors;
  - avoiding the routing of high-activity pipes through low-radiation zones;
  - use of shielded pipe trenches and pipe chases, where routing of high-activity pipes in low-level areas cannot be avoided.

– separating radioactive and non-radioactive pipes for maintenance purposes.

- To maintain acceptable levels at the valve stations, motor-operated or diaphragm valves are used where practical. For valve maintenance, provision is made for draining and flushing associated equipment so that radiation exposure is minimized. If manual valves are used, provision is made for shielding the operator from the valve by use of shield walls and valve stem extensions, where practicable.

- Shielding is provided to permit access and occupancy of the control room to ensure that plant personnel exposure following an accident does not exceed the guideline values set forth in 10 CFR 50, Appendix A, Criterion 19. The analyses of the doses to Control Room personnel for the design basis accidents are included in Chapter 15.

- The dose at the site boundary as a result of direct and scattered radiation from the turbine and associated equipment is considered.

- In selected situations, provisions are made for shielding major radiation sources during inservice inspection to reduce exposure to inspection personnel. For example, steel platforms are provided for ISI of the RPV nozzle welds and associated piping.

- The primary material used for shielding is concrete at a density of 2.35 gm/cm^3. Concrete used for shielding purposes is designed in accordance with Regulatory Guide 1.69. Where special circumstances dictate, steel, lead, water, lead-loaded silicone foam, or a boron-laced refractory material is used.

12.3.2.2.2 Method of Shielding Design

The radiation shield wall thicknesses are determined using basic shielding data and proven shielding codes. A list of the computer programs used is contained in Table 12.3-1. The shielding design methods used also rely on basic radiation transport equations contained in Reference 12.3-1. The sources for basic shielding data, such as cross sections, buildup factors, and radioisotope decay information, are listed in References 12.3-2 through 12.3-10.

The shielding design is based on the plant operating at maximum design power with the release of fission products resulting in a source of noble gas after a 30-minute decay period, and the corresponding activation and corrosion product concentrations in the reactor water listed in Section 11.1. Radiation sources in various pieces of plant equipment are cited in Section 12.2. Shutdown conditions, such as fuel transfer operation, as well as accident conditions, such as a Loss of Coolant Accident (LOCA) or a Fuel Handling Accident (FHA), have also been considered in designing shielding for the plant.

The mathematical models used to represent a radiation source and associate equipment and shielding are established to ensure conservative calculation results. Depending on the versatility of the applicable computer program, various degrees of complexity of the actual physical situation are incorporated. In general, cylindrically shaped equipment such as tanks, heat exchangers, and demineralizers are mathematically modeled as truncated cylinders. Equipment internals are sectional and homogenized to incorporate density variations where applicable. For example, the tube bundle section of a heat exchanger exhibits a higher density than the tube bundle clearance circle, due to the tube density, and this variation is accounted for in the model. Complex piping runs are conservatively modeled as a series of point sources spaced along the piping run. Equipment containing sources in a parallelepiped configuration, such as fuel assemblies and fuel racks are modeled as parallelepiped with a suitable homogenization of materials contained in the equipment. The shielding for these sources is also modeled on a conservative basis, with discontinuities in the shielding, such as penetrations, doors, and partial walls accounted for. The dimension of the floor decking is not considered in the shielding calculation as it is part of the effective shield thickness provided by the floor slab.

Pure gamma dose rate calculations, both scattered and direct, are conducted using point kernel codes (QADF/GGG). The source terms are divided into groups as a function of photon energy, and each group is treated independently of the others. Credit is taken for attenuation through all phases of material, and buildup is accounted for using a third-order polynomial buildup factor equation. The more conservative material buildup coefficients are selected for laminated shield configuration to ensure conservative results.

For combined gamma and neutron shielding situations, discrete ordinates techniques (DORT) are applied.

The shielding thicknesses are selected to reduce the aggregate dose rate from significant radiation sources in surrounding areas to values below the upper limit of the radiation zone specified in the zone maps in Subsection 12.3.1.3. By maintaining dose rates in these areas at less than the upper limit values specified in the zone maps, sufficient access to the plant areas is allowed for maintenance and operational requirements.

Where shielded entries to high-radiation areas such as labyrinths are required, a gamma ray scattering code (GGG) is used to confirm the adequacy of the labyrinth design. The labyrinths are designed to reduce the scattered as well as the direct contribution to the aggregate dose rate outside the entry, such that the radiation zone designated for the area is not violated.

12.3.2.2.3 Plant Shielding Description

The general description of the shielding is provided below:

Containment - The major shielding structures located in the drywell area consist of the reactor shield wall and the drywell wall. The reactor shield wall in general consists of 16 cm of steel plate. The primary function served by the reactor shield wall is the reduction of radiation levels in the drywell due to the reactor, to valves that do not unduly limit the service life of the equipment located in the drywell. In addition, the reactor shield wall reduces gamma heating effects on the drywell wall, as well as providing for low radiation levels in the drywell during reactor shutdown. The drywell is an F radiation zone during full power reactor operation and is not accessible during this period.

The containment (drywell) outside wall is a 2 m thick reinforced concrete cylinder that totally surrounds the drywell. A 2.4m thick reinforced concrete containment top slab tops the drywell. The drywell wall attenuates radiation from the reactor and other radiation sources in the drywell to allow occupancy of the reactor building during full power reactor operation.

The ESBWR plant includes all necessary shielding provisions in the upper drywell in order to reduce the dose ALARA during transfer of irradiate spent fuel assemblies. In such a way ESBWR plant includes all applicable shielding design provisions to minimize dose rates in case of fuel handling mishap resulting in dropping a fuel assembly across the reactor flange.

Reactor Building - In general, the shielding for the reactor building is designed to maintain open areas at dose rates less than 6 µSv/hr (0.6 mR/hr).

Penetrations of the containment wall are shielded to reduce radiation streaming through the penetrations. Localized dose rates outside these penetrations are limited to less than 50 µSv/hr (5 mR/hr). The penetrations through interior shield walls of the reactor building are shielded using a lead-loaded silicone sleeve to reduce the radiation streaming. Penetrations are also located so as to minimize the consequences of radiation streaming into surrounding areas.

The components of the RWCU/SDC are located in the reactor building. Both the RWCU regenerative and nonregenerative heat exchangers are located in shielded cubicles separated from the other components of the system. Neither cubicle needs to be entered for system operation.

Process piping between the heat exchangers and the demineralizers is routed through shielded areas or embedded in concrete to reduce the dose rate in surrounding areas. The RWCU/SDC demineralizers are located in separate shielded cubicles. This arrangement allows maintenance of one unit while operating the other. The dose rate in the adjoining demineralizer cubicle from the operating unit is less than 60 µSv/hr (6 mR/hr). Entry into the demineralizer cubicle, which is required infrequently, is via a labyrinth entryway. The bulk of the piping and valves for the filter demineralizers is located in an adjacent shielded valve gallery. Backfilling and resin application of the filter demineralizers are controlled from an area where dose rates are less than 10 µSv/hr (1 mR/hr).

The ESBWR employs a passive cooling system in addition to the RWCU/SDC for cooling the core and vessel. Access into the cubicles is not required to operate the systems. All such components that could become contaminated in the event of an accident are located in the containment except those components that would be used as part of the RWCU/SDC.

Fuel Storage - The fuel storage pool is designed to ensure the dose rate around the pool area is less than 25 µSv/hr (2.5 mR/hr). In the event of an anticipated operational occurrence where the fuel sustains significant damage, such as a fuel drop accident, airborne dose rates in the pool area may significantly exceed this dose rate.

Control Room - The dose rate in the control room is limited to 6 µGy/h during normal reactor operating conditions. The outer walls of the Control Building are designed to attenuate radiation from radioactive materials contained within the Reactor Building and from possible airborne radiation surrounding the Control Building following a LOCA. The walls provide sufficient shielding to limit the direct-shine exposure of control room personnel following a LOCA to a fraction of the 5 Rem limit as is required by 10 CFR 50 Appendix A, Criterion 19.

Main steam tunnel - The main steam tunnel extends from the primary containment boundary in the Reactor Building up to the turbine stop valves. The primary purpose of the steam tunnel is to shield the plant complex from N-16 gamma shine in the main steam lines. The tunnel walls provide sufficient shielding to limit the direct-shine exposure from the main steam lines in any point that may be inhabited during normal operations.

12.3.3 Ventilation

The HVAC systems for the various buildings in the plant are discussed in Section 9.4, including the design bases, system descriptions, and evaluations with regard to the heating, cooling, and ventilating capabilities of the systems. This Subsection discusses the radiation control aspects of the HVAC systems.

12.3.3.1 Design Objectives

The following design objectives apply to all building ventilation systems:

- The systems shall be designed to make airborne radiation exposures to plant personnel and releases to the environment ALARA. To achieve this objective, the guidance provided in Regulatory Guide 8.8 shall be followed.
- The concentration of radionuclides in the air in areas accessible to personnel for normal plant surveillance and maintenance shall be kept below the limits of 10 CFR 20 during normal power operation. This is accomplished by establishing in each area a reasonable compromise between specifications on potential airborne leakages in the area and HVAC flow through the area. Appendix 12.A to this chapter outlines the methodology by which such calculations are made.

The COL licensee will perform calculations for the expected airborne radionuclide concentrations to verify the adequacy of the ventilation system prior to fuel load. See Subsection 12.3.7 for COL license information .

12.3.3.2 Design Description

In the following subsections, the design features of the various ventilation systems that achieve the radiation control design objectives are discussed. For all areas potentially having airborne radioactivity, the ventilation systems are designed such that during normal and maintenance operations, airflow between areas is always from an area of low potential contamination to an area of higher potential contamination.

12.3.3.2.1 Control Room Ventilation

The control building atmosphere is maintained at a slightly positive pressure (up to 0.5 in. Hg) at all times in order to prevent infiltration of contaminants. When offsite power is available, fresh air may be taken in via the single inlet system, which has its intake structure on the side of the building. During an isolation event if offsite or backup power is not available, bottled air can be supplied by a redundant supply system for up to 72 hours prior to requiring recharging. Under conditions when offsite or backup power is available, either bottled or filtered air may be used. The operator has manual control in the event filtered air is used to either run under filtered air or bottled air.

Outside a particulate filter normally filters air that enters the intake. Under contamination conditions, however, if external air is selected, the airflow is diverted through an adsorber system having:

- a particulate filter;
- a HEPA filter;
- a charcoal filter; and
- another HEPA filter.

The outdoor cleanup unit is located in a closed room that helps prevent the spread of any radiation during maintenance. Adequate space is provided for maintenance activities. The particulate and HEPA filters can be bagged when being removed from the unit. Before removing the charcoal, any radioactivity is allowed to decay to minimal levels, and is then removed through a connection in the bottom of the filter by a pneumatic transfer system. Air used in the transfer system goes through a HEPA filter before being exhausted. Facemasks can be worn during maintenance activities, if desired.

For a complete description of the control room HVAC system see Subsection 9.4.1.

12.3.3.2.2 Containment

Access into the containment drywell is not permitted during normal operation. The ventilation system inside merely circulates, without filtering, the air. The only airflow out of the drywell into accessible areas is minor leakage through the wall. During maintenance, the drywell air is purged before access is allowed.

12.3.3.2.3 Reactor Building

The reactor building HVAC system is divided into two major components: the contaminated and the clean areas. The clean area system conditions and circulates air through all the clean areas of the reactor building. The contaminated area system conditions and circulates air through the contaminated areas of the building. Flow into both areas is directed from the corridors (point of highest pressure) to the equipment alcove rooms, then to the rooms themselves, and finally to the external wall pipe chases and from the pipe chases back to the HVAC system. The clean area system dumps circulated air to the environment through building vents, while the contaminated air system directs flow through the HVAC system to the plant stack. Under isolation conditions, the HVAC system isolates to localize any contamination until operator action determines the best method for decontamination.

For a description of the reactor building HVAC system, see Subsection 9.4.6.

12.3.3.2.4 Radwaste Building

The radwaste building is divided into two zones for ventilation purposes. The control room is one zone, and the remainder of the building is the other zone. The air pressure in the first zone is maintained slightly above atmospheric, while the air pressure in the second zone is maintained slightly below atmospheric. Air in the second zone is drawn from outside the building and distributed to various work areas within the building. Air flows from the work areas and is then discharged via the reactor building stack. An alarm sounds in the control room if the exhaust fan fails. The exhaust flow is monitored for radioactivity, and if a high activity level is detected, the potentially radioactive cells are automatically isolated, but airflow through the work areas continues.

If the exhaust flow high-radiation alarm continues to annunciate after the tank and pump rooms are isolated, the work area branch exhaust ducts are selectively manually isolated to locate the involved building area. Should this technique fail, because the airborne radiation has spread throughout the building, the control room air conditioning continues, but the air conditioning for the balance of the building is shut down.

The work area's exhaust air is drawn through a filter unit consisting of a particulate filter and a HEPA filter before being discharged to the reactor building stack. The air is monitored for radioactivity, and if a high level is detected, supply and exhaust is terminated.

Maintenance provisions for the filters are similar to those for the control building HVAC System.

See Subsection 9.4.3 for a detailed discussion of the radwaste building HVAC System.

12.3.3.2.5 Fuel Building

The Fuel Building Heating, Ventilation, and Air Conditioning (FBHV) systems are the Fuel Building General Area HVAC (FBGAHV) and Fuel Building Fuel Pool Area HVAC (FBFPHV) subsystems. The FBGAHV serves the general area. The FBFPHV serves the refueling floor and pool areas. The FBHV systems operate during normal plant operation, plant startup, and plant shutdown.

The FBGAHV consists of two 100% capacity AHUs with two 100% capacity supply fans, two 100% capacity exhaust fans, recirculation AHUs, and unit heaters. The FBGAHV incorporates a common supply and return duct system that distributes conditioned air to the general area of the Fuel Building and exhaust air to the outside atmosphere. During normal operation, air travels through the AHU's stages where particulates are removed from the air by low and high efficiency filters; heat is transferred between the mixed air and the hot/chilled water coils; and the conditioned air is distributed to the clean areas by the supply fan. Exhaust air is ducted to the exhaust fan and exhausted to the outside atmosphere.

The FBFPHV consists of two 100% capacity AHUs with two 100% capacity supply fans, two 100% capacity exhaust fans, and redundant bubble-tight isolation dampers. The FBFPHV is a once-through ventilation system that distributes conditioned air to the refueling area of the reactor and spent fuel pool area of the Fuel Building. During normal operation, outside air travels through the AHU's stages where particulates are removed from the air by low and high efficiency filters; heat is transferred between the air and the hot/chilled water coils; and the conditioned air is distributed to the refueling area and spent fuel pool surfaces. Air is ducted to the exhaust fan and exhausted to the outside atmosphere through the plant vent stack. The exhaust system has the manual capability to divert the exhaust for filtration by the purge exhaust filter unit, prior to discharge to the plant vent stack. FBFPHV exhaust fans are used for smoke removal.

The common plant vent stack provides monitoring and discharging of FBGAHV and FBFPHV exhausts. See Subsection 9.4.2 for a detailed discussion of the FBHV system.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The following systems are provided to monitor area radiation and airborne radioactivity within the plant:

- The Area Radiation Monitoring System (ARMS) continuously measures, indicates and records the gamma radiation levels at strategic locations throughout the plant except within the primary containment, and activates alarms in the main control room as well as in local areas to warn operating personnel to avoid unnecessary or inadvertent exposure to radiation. This system is classified as nonsafety-related.

- The Containment Monitoring System (CMS) continuously measures, indicates, and records the gamma radiation levels within the primary containment (drywell and suppression chamber), and activates alarms in the main control room on high radiation level. As described in Subsection 7.5.2, four gamma sensitive ion chambers are provided within the primary containment to monitor gamma rays during normal, abnormal and accident conditions. Two redundant sensors are located in the drywell and two in the wetwell. The range of each monitor covers 7 decades from 0.01 Gy/hr (1R/hr) to 10^5 Gy/hr (10^7 R/hr) as required by RG 1.97. The CMS is classified as safety-related
- Airborne radioactivity in effluent releases and ventilation air exhausts is continuously sampled and monitored by the Process Radiation Monitoring System (PRMS) for noble gases, air particulates and halogens. As described in Section 11.5, airborne contamination is sampled and monitored at the stack common discharge, in the off-gas releases, and in the ventilation exhaust from the reactor, radwaste and turbine buildings. Samples are periodically collected and analyzed for radioactivity. In addition to this instrumentation, portable air samplers are used for compliance with 10 CFR 20 restrictions to check for airborne radioactivity in work areas prior to entry where potential radiation levels may exit that exceed the allowable limits.
- The radiation instrumentation that monitors airborne radioactivity is classified as nonsafety-related, and the COL applicant will provide detailed information. See Subsection 12.3.7 for COL license information.

12.3.4.1 ARM System Description

The ARMS consists of gamma sensitive detectors, digital area radiation monitors, and local auxiliary units with indicators and local audible alarms. The output signals from the detectors are digitized and multiplexed for transmission to digital radiation monitors for measurement and display. Also, the radiation signals are transmitted to the process computer for recording. Each radiation monitoring channel has two adjustable trip alarm circuits, one for high radiation and the other for downscale indication (loss of sensor input). Also, each area radiation monitor has a built-in self test capability that checks for gross failures and activates an alarm on power failure or inoperative monitor. Auxiliary units with local audible alarms are provided in selected local areas to provide for immediate warning in order to minimize occupational exposure. Each area radiation monitor is powered from non-1E vital 120 VAC power source, which is continuously available during loss of off-site power.

12.3.4.2 ARM Detector Location and Sensitivity

The detector locations are shown on plant layout drawings for each building (Figures 12.3-23 through 12.3-42). The area radiation channels for each building are listed in Tables 12.3-2 through 12.3-6, along with reference to the figure that shows the detector location, the channel monitoring range, and the local area alarms assignment. The monitoring range of each area radiation channel is shown in Table 12.3-7.

12.3.4.3 Pertinent Design Parameters and Requirements

Two high-range radiation channels are provided in the fuel transfer and storage area to monitor radiation that may result from a fuel handling accident. Criticality detection monitors are not needed to satisfy the criticality accident requirements of 10 CFR 70.24 because the ESBWR utilizes high-density fuel storage racks that are designed to be subcritical under normal, abnormal and accident conditions. The new fuel bundles are stored under water in storage racks that are located in the fuel vault adjacent to the reactor cavity, while the spent fuel bundles are stored in racks that are placed at the bottom of the fuel storage pool. A full array of loaded new or spent fuel storage racks is designed to be subcritical as defined in Subsections 9.1.1 and 9.1.2, respectively.

The detectors and radiation monitors are responsive to gamma radiation over an energy range of 80 keV to 7 MeV. The energy dependence does not exceed 20% of point from 100keV to 3 MeV. The overall system design accuracy is within 10% of equivalent linear full-scale output for any decade.

The alarm setpoints are established in the field following equipment installation at the site. The exact settings are based on sensor location, background radiation levels, expected radiation levels, and low occupational radiation exposures. The radiation alarm setpoint for each channel is set slightly above the background radiation level that is normal in the area where the monitor is located.

Each channel is calibrated based on a pseudo input signal to verify monitor response. Each detector is calibrated using a radioactive source traceable to the National Institute of Standards and Technology. The area radiation monitors are checked and calibrated periodically.

The ARMS is designed to provide early detection and warning for personnel protection to ensure occupational radiation exposures are as low as reasonably achievable (ALARA) in accordance with guidelines stipulated in RG 8.2 and RG 8.8. Also, the ARMS includes instrumentation in crucial areas of the reactor building where access may be required to service safety-related equipment following a LOCA event.

12.3.5 Post-Accident Access Requirements

The locations requiring access to mitigate the consequences of an accident during the 100-day post-accident period are the main control room, the technical support center, the remote shutdown panel, the primary containment sample station (PASS), the health physics facility (counting room), the control room air bottles (EBAS), the isolation condenser (IC) pool refill nozzles, and the nitrogen gas supply bottles. Each area has low post LOCA radiation levels. The dose evaluations in Subsection 15.3.3 are within regulatory guidelines.

Access to vital areas through out the reactor building/control building/turbine building complex is controlled via the service building. Entrance to the service building and access to the other areas are controlled via double locked secured entryways. Access to the reactor building is via two specific routes, one for clean access and the second for controlled access. During an event such as a design basis accident, the control building is maintained under filtered HVAC at positive pressure with respect to the environment; otherwise, the main control room is maintained at positive pressure from the bottled air supply (EBAS). Air infiltration is minimized by positive flow via double entry ways. Therefore, radiation exposure is limited to gamma shine from the reactor building, turbine building, main steam line access corridor, and skyline.

During a design basis accident event, access to nitrogen bottles, the PASS, and monitor systems is controlled from the service building via the controlled access way. These corridors are not maintained under filtered positive pressure so personal protection equipment (radiation protection suits, breathing gear, etc.) is required in the access corridor. Primary contamination would occur from leakage through the PASS and air infiltration from the environment. Both pathways are considered minimal and minor contamination under even the most adverse conditions is expected. Access to the IC pool refill nozzles and the control room air bottles is from outside and would not likely require special breathing gear, etc.

The reactor building vital areas are all located off the controlled access way and contamination is limited to air infiltration from the accident environment and penetration leakage from the PASS. Sources of radiation in each area are limited to gamma shine from the reactor building and potential leakage from monitor systems such as the PASS. These sources are considered minimal including the stack monitor room which contains only instrumentation with their associated penetrations for monitoring stack effluent.

12.3.6 Post-Accident Radiation Zone Maps

The post-accident radiation zone maps for the areas in the reactor building are presented in Figures 12.3-43 through 12.3-51. The zone maps represent the maximum gamma dose rates that exist in these areas during the post-accident period. These dose rates do not include the airborne contribution in the reactor building. The zone maps are designed to reflect the criteria established in Subsection 3.1.2.

12.3.7 COL Information

12.3.7.1 Facility Design Features

Precise equipment definition and material selection are the COL applicant's responsibility.

12.3.7.2 Airborne Radionuclide Concentration Calculation

The COL applicant will provide the calculations of the expected concentrations of the airborne radionuclide for the ESBWR plant design (Subsection 12.3.3.1).

12.3.7.3 Operational Considerations

Area radiation monitoring operational considerations, such as monitor alarm setpoints, listed in Regulation Guide 1.70 are the COL applicant's responsibility. Airborne radiation monitoring operational considerations such as the procedures for operations and calibration of the monitors, as well as the placement of the portable monitors, are also the COL applicant's responsibility (Subsection 12.3.4).

12.3.8 References

12.3-1 U.S. Atomic Energy Commission, "Reactor Shielding for Nuclear Engineers," TID-25951, 1973.

12.3-2 U.S. Department of Commerce, "Photon Cross Sections, Attenuation Coefficients, and Energy Absorption Coefficients from 10 KeV to 100 GeV," NSRDS-NBS20, August 1969.

12.3-3 U.S. Department of Health, Education, and Welfare, "Radiological Health Handbook," Revised Edition, January 1970.

12.3-4 U.S. Atomic Energy Commission, "Reactor Handbook, Volume III, Part B," 1962.

12.3-5 Lederer, Hollander, and Perlman, "Table of Isotopes," Sixth Edition (1968).

12.3-6 General Electric Company, "Polynomial Approximation of Gamma Ray Buildup Factors for a Point Isotropic Source", APEX-510, November 1958.

12.3-7 U.S. Atomic Energy Commission, "Reactor Physics Constants, Second Edition," ANL-5800, July 1963.

12.3-8 Brookhaven National Laboratory, "ENDF/B-III and ENDF/B-IV Cross Section Libraries"

12.3-9 Oak Ridge National Laboratory, "PDS-31 Cross Section Library"

12.3-10 "DLC-7, ENDF/B Photo Interaction Library"

Table 12.3-1

Computer Programs Used in Shielding Design Calculations

Computer Code	Description
QADF	A multigroup, multiregion, point kernel gamma radiation code for calculating the flux and dose rate at discrete locations within a complex source geometry configuration.
GGG	A multigroup, multiregion, point kernel code for calculating the contributions due to gamma ray scattering in a heterogeneous three-dimensional space.
DORT	A discrete ordinates two-dimensional transport code. Multigroup, multiregion neutron or gamma transport.
QAD CGGP 1.0	"Quick and Dirty Combinatorial Geometry –Geometric Progression". A multigroup, multiregion, point kernel gamma radiation code for calculating the flux and dose rate at discrete locations within a complex source geometry configuration

Table 12.3-2

Area Radiation Monitors for Reactor Building

No. [1]	Description & Location	Figure No.	Monitoring Range	Aux. Units
1.	Refueling Floor Area #1, EL 34000	12.3-31	H	
2.	Refueling Floor Area # 2, El 34000	12.3-31	H	
3.	New Fuel Storage Pool, EL 27000	12.3-30	H	
4.	New Fuel Storage Pool, EL 27000	12.3-30	H	
17.	RWCU/SDC Pump, EL -11500	12.3-23	H	
18	RB Sump Pumps, EL -11500	12.3-23	H	
19.	RWCU/SDC Train A Heat Exchanger EL -11500	12.3-23	H	
20.	RWCU/SDC Train B Heat Exchanger EL -11500	12.3-23	H	
21.	Equipment Hatch Pathway, EL -6400	12.3-24	M	
22.	Personnel Hatch Pathway, EL -6400	12.3-24	H	
23	FMCRD HCU Area #1, EL -6400	12.3-24	M	
25.	FMCRD HCU Area # 3, EL -6400	12.3-24	M	
27.	RWCU/SDC Filter Demineralizer Area (Near Equip. Hatch), EL -1000	12.3-25	H	
28.	Radiological Control Area Entrance, EL 17500	12.3-29	M	
29	Hydrogen/Oxygen Monitoring (CMS), Skid EL 13570	12.3-28	H	
30.	Hydrogen/Oxygen Monitoring (CMS), Skid EL 13570	12.3-28	H	
31	Instrument Rack Area #1, EL -11500	12.3-23	H	
32.	Instrument Rack Area #2, EL -11500	12.3-23	H	
33	Instrument Rack Area #3, EL -11500	12.3-23	H	
34	Instrument Rack Area #4, EL -11500	12.3-23	H	
35	Instrument Rack Area #5, EL -11500	12.3-23	H	
36	Instrument Rack Area #6, EL -11500	12.3-23	H	
37	Instrument Rack Area #7, EL -11500	12.3-23	H	
38	Instrument Rack Area #8, EL -11500	12.3-23	H	
39	Fuel Transfer System (FTS) Maintenance Room (Multiple), EL 17500	12.3-29	H	X
40	Fuel Handling Machine (IFTS), EL 34000	12.3-31	H	
41	Remote Shutdown Panel A Area, El. –1000	12.3-25	H	
42	Remote Shutdown Panel B Area, El. –1000	12.3-25	H	

[1] Notes #5 through 16, 24 and 26 not used

Table 12.3-3

Area Radiation Monitors for Fuel Building

No.[2]	Description & Location	Figure No.	Monitoring Range	Aux. Units
1.	Spent Fuel Floor, EL 4650	12.3-26	H	
2.	Fuel Handling Machine, EL 4650	12.3-26	M	
3.	Fuel Transfer Cask Area, EL 4650	12.3-26	H	
5	FAPCS Heat Exchangers, EL -11500	12.3-23	H	
6	FAPCS System Transfer Pumps, EL -11500	12.3-23	H	
9	Sump Pumps, EL -11500 H	12.3-23	H	
10	Ground Grade Access Pathway, EL 4650	12.3-26	M	
11	Wash Down Bay Entry Door, EL 4650 (Truck)	12.3-26	H	
12	Fuel Transfer System (FTS) Maintenance Rooms (Multiple) EL 4650	12.3-26	H	

[2] Notes #4, 7 & 8 not used

Table 12.3-4

Area Radiation Monitors for Radwaste Building

No.	Description & Location	Figure No.	Monitoring Range	Aux. Units
1.	Electrical Board Room El -9350	12.3-39	H	
2.	Control Room	12.3-39	H	
3.	High Activity Resin Recirculation Pump Room, El –9350	12.3-39	H	
4.	High Activity Resin Transfer Pump Room, El –2350	12.3-39	H	
5.	Trailer Access Area El 4650	12.3-41	H	
6.	Liquid Radioactive Waste Treatment Area (Hallover Fiber deep-Bed Demineralizer, Reverse Osmosis System, etc.) El 4650	12.3-41	H	
7.	Wet Solid Radioactive Waste Treatment Area (Dewatering Equipment, Concentrate Treatment System, etc.) EL4650	12.3-41	H	
8.	Dry Solid Waste Treatment Area (High Dose Rate Waste Storage Area, etc.) El 4650	12.3-41	H	
9.	Packaged Waste Staging Area, El 4650	12.3-41	H	

Table 12.3-5

Area Radiation Monitors for Turbine Building

No.[3]	Description & Location	Figure No.	Monitoring Range	Aux. Units
1.	Main Condenser Floor Area EL -1400	12.3-32	M	X
2.	Drain Cooler Area EL 4650	12.3-33	M	
3.	Offgas Sampling Area EL 4650	12.3-33	M	X
4.	Condensate Pumps Area EL -1400	12.3-32	M	
5.	Low Pressure Heater Area EL 20000	12.3-35	M	
6	Deaerator Area, EL 28000	12.3-36	M	
7	SRV/MSIV Maintenance Area EL 20000	12.3-35	M	
8	Steam Jet Air Ejector (SJAE) B Area EL 4650	12.3-33	M	X
9	SJAE A Area EL 4650	12.3-33	M	
10	High Pressure Heater Area EL 20000	12.3-35	M	
11	Filters and Demineralizers Area EL 4650	12.3-34	M	X
12	Turbine Operating Floor Area EL 28000	12.3-36	M	X
13.	Turbine Operating Floor Area EL 28000	12.3-36	M	
14	Crane Travel Area (Various)	12.3-38	M	X
15	Equipment Main Access Area, EL 4650	12.3-33	M	
16.	RCCW System Area Entrance EL 4650	12.3-33	M	
17	Offgas Charcoal Adsorber Room Entrance Area EL -1400	12.3-32	M	
18	Backwash Transfer [4]Pumps Entrance Area EL -1400	12.3-32	M	
19	Condensate Hollow Fiber Filter Valve Room EL -1400	12.3-32	M	
20	Sample Room Area EL -1400	12.3-32	M	
22	Condensate D/B Demineralizer Entrance Area, EL 4650	12.3-33		
23	Offgas Hydrogen Recombiner A, EL 12000	12.3-34	M	
24	Offgas Hydrogen Recombiner B, EL 4650	12.3-33	M	
25	Instrument Air Compressor Area, EL 12000	12.3-34	M	
26	MCC Water Chiller Room A, EL 28000	12.3-36	M	
27	MCC Water Chiller Room B, EL 28000	12.3-36	M	
28	Turbine Building Exhaust Duct Area EL 33000	12.3-37	M	
29	RCCWS Area Entrance EL 4650	12.3-33	M	

[3] Note #21 not used

Table 12.3-6

Area Radiation Monitors for Control Building

No.	Description & Location	Figure No.	Monitoring Range	Local Alarms
1.	Main Control Room, EL -1000	12.3-25	H	

Table 12.3-7

Area Radiation Channel Monitoring Range

Low Setting	High Setting	Descriptor
1E-4 mSv/h; 1E-4 mGy/hr (1E-5 R/hr)	1E0 mSv/h, 1E0 mGy/hr (1E-1 R/hr)	H (High Sensitivity)
1E-3 mSv/h; 1E0 μGy/hr (1E-4 mR/hr)	1E1 mSv/h; 1E1 mGy/hr (1E0 R/hr)	M (Medium Sensitivity)
1E-2 mSv/h; 1E1 μGy/hr (1E0 mR/hr)	1E2 mSv/h; 1E2 mGy/hr (1E1 R/hr)	L (Low Sensitivity)
1E0 mSv/h; 1E0 mGy/hr (1E2 mR/hr)	1E4 mSv/h; 1E1 Gy/hr (1E6 mR/hr)	LL (Low-Low Sensitivity)
1E-4 Sv/h; 1E-1 mGy/hr (1E-2 R/hr)	1E2 Sv/h; 1E2 Gy/hr (1E4 R/hr)	VL (Very Low Sensitivity)

Figure 12.3-1. Nuclear Island Radiation Zones for Full Power and Shutdown Operation at Elevation -11500 mm[EA81]

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-2. Nuclear Island Radiation Zones for Full Power and Shutdown Operation at Elevation -6400 mm[EA82]

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Figure 12.3-3. Nuclear Island Radiation Zones for Full Power and Shutdown Operation at Elevation -1000 mm[EA83]

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Figure 12.3-4. Nuclear Island Radiation Zones for Full Power and Shutdown Operation at Elevation 4650 mm[EA84]

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Figure 12.3-5. Nuclear Island Radiation Zones for Full Power and Shutdown Operation at Elevation 9060 mm[EA85]

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-6. Nuclear Island Radiation Zones for Full Power and Shutdown Operation at Elevation 13570 mm[EA86]

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Figure 12.3-7. Nuclear Island Radiation Zones for Full Power and Shutdown Operation at Elevation 17500 mm[EA87]

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Figure 12.3-8. Nuclear Island Radiation Zones for Full Power and Shutdown Operation at Elevation 27000 mm[EA88]

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Figure 12.3-9. Nuclear Island Radiation Zones for Full Power and Shutdown Operation at Elevation 34000 mm[EA89]

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Figure 12.3-10. Nuclear Island Radiation Zones for Full Power and Shutdown Operation Section A-A[EA90]

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Figure 12.3-11. Nuclear Island Radiation Zones for Full Power and Shutdown Operation Section B-B[EA91]

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Figure 12.3-12. Turbine Building Radiation Zones at Elevation -1400 mm[EA92]

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Figure 12.3-13. Turbine Building Radiation Zones at Elevation 4650 mm[EA93]

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-14. Turbine Building Radiation Zones at Elevation 12000 mm

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Figure 12.3-15. Turbine Building Radiation Zones at Elevation 20000 mm[EA95]

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Figure 12.3-16. Turbine Building Radiation Zones at Elevation 28000 mm[EA96]

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Figure 12.3-17. Turbine Building Radiation Zones at Elevations 33000 and 38000 mm[EA97]

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Figure 12.3-18. Turbine Building Radiation Zones at Elevation Various[EA98]

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-19. Radwaste Building Radiation Zones at Elevation -9350 mm [EA99]

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Figure 12.3-20. Radwaste Building Radiation Zones at Elevation -2350 mm[EA100]

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Figure 12.3-21. Radwaste Building Radiation Zones at Elevation 4650 mm [EA101]

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Figure 12.3-22. Radwaste Building Radiation Zones at Elevation 10650 mm [EA102]

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Figure 12.3-23. Nuclear Island Area Radiation Monitors at Elevation -11500 mm[EA103]

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Figure 12.3-24. Nuclear Island Area Radiation Monitors at Elevation -6400 mm

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-25. Nuclear Island Area Radiation Monitors at Elevation -1000 mm

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-26. Nuclear Island Area Radiation Monitors at Elevation 4650 mm

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Figure 12.3-27. Nuclear Island Area Radiation Monitors at Elevation 9060 mm

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-28. Nuclear Island Area Radiation Monitors at Elevation 13570 mm

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-29. Nuclear Island Area Radiation Monitors at Elevation 17500 mm

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-30. Nuclear Island Area Radiation Monitors at Elevation 27000 mm

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Figure 12.3-31. Nuclear Island Area Radiation Monitors at Elevation 34000 mm

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Figure 12.3-32. Turbine Building Area Radiation Monitors at Elevation -1400 mm

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-33. Turbine Building Area Radiation Monitors at Elevation 4650 mm

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Figure 12.3-34. Turbine Building Area Radiation Monitors at Elevation 12000 mm

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Figure 12.3-35. Turbine Building Area Radiation Monitors at Elevation 20000 mm

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-36. Turbine Building Area Radiation Monitors at Elevation 28000 mm

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-37. Turbine Building Area Radiation Monitors at Elevation 33000 and 38000 mm

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-38. Turbine Building Area Radiation Monitors at Elevation Various

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-39. Radwaste Building Area Radiation Monitors at Elevation -9350 mm

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-40. Radwaste Building Area Radiation Monitors at Elevation -2350 mm

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-41. Radwaste Building Area Radiation Monitors at Elevation 4650 mm

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Figure 12.3-42. Radwaste Building Area Radiation Monitors at Elevation 10650 mm

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-43. Nuclear Island Post Accident Radiation Zones at Elevation -11500 mm

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Figure 12.3-44. Nuclear Island Post Accident Radiation Zones at Elevation -6400 mm

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-45. Nuclear Island Post Accident Radiation Zones at Elevation -1000 mm

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-46. Nuclear Island Post Accident Radiation Zones at Elevation 4650 mm

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-47. Nuclear Island Post Accident Radiation Zones at Elevation 9060 mm

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-48. Nuclear Island Post Accident Radiation Zones at Elevation 13570 mm

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-49. Nuclear Island Post Accident Radiation Zones at Elevation 17500 mm

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-50. Nuclear Island Post Accident Radiation Zones at Elevation 27000 mm

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390}}}

Figure 12.3-51. Nuclear Island Post Accident Radiation Zones at Elevation 34000 mm

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12.4 DOSE ASSESSMENT

This section discusses the radiological dose assessment (person-Sievert) for the ESBWR facility. Subsections 12.4.1 through 12.4.5 discuss the various factors involved with dose assessment within the different plant radiation areas. The resulting annual radiation dose estimate is summarized in Table 12.4-1.

Dose assessment is a significant element in determining that the facility design and methods of operation ensure occupational radiological exposures are as low as reasonably achievable (ALARA). Dose assessment depends on estimates of occupancy, dose rates in various occupied areas, frequency of operations, and number of personnel participating in reactor operations and surveillance, routine maintenance, waste processing, refueling, in-service inspection and special (unscheduled) maintenance. Facility personnel include station and utility employees, as well as contract workers. The occupations for these personnel include maintenance, operations, health physics, supervision and engineering.

To estimate the total annual radiological dose to personnel, five facility area/task designations are specified, and the annual person-Sievert dose for each area/task is evaluated separately. These designations are listed in Table 12.4-1. Where job functions and expected radiation levels are predictable or clearly defined, analytical methods were employed for the person-Sievert estimates. The resulting dose estimates for the ESBWR are contained in Table 12.4-1. Subsections 12.4.1 through 12.4.6 discuss the various factors involved in the evaluations and their related elements.

The analytical method used for the person-Sievert assessment is based on the product of the estimated occupancy time (i.e., person-hours per year) and the estimated average dose rate. Estimates of the occupancy time required for operations associated with equipment in facility radiation areas (e.g., maintenance, testing or surveillance time) are first determined. An applicable frequency of occurrence is also incorporated in the resulting annual occupancy time for each operation. Areas with insignificant radiation sources or occupancy are not included in the exposure estimate. Where radiation sources are present, a design maximum dose rate of 10 μSv/hr (1 mrem/hr) is assumed for Radiation Zone B areas and 50 μSv/hr (5 mrem/hr) for Radiation Zone C areas. Other estimated dose rates are based on either calculations or extrapolated from radiation levels reported at operating plants.

The primary purpose for dose assessment is to aid in reducing the radiological exposure associated with all phases of plant operation consistent with practical considerations for performing each task. To achieve this ALARA objective, the ESBWR design includes numerous significant design improvements to reduce occupational exposures from past BWR experience. For example, facility design improvements include the elimination of recirculation piping and valves, improved water chemistry and low cobalt alloys for the reactor cooling water boundary, a simplified Control Rod Drive (CRD) system, reduced equipment maintenance and improved access, increased use of live-load valve packing to mitigate stem leakage, overhaul handling and refueling devices, multiple main steam line plugs, automatic MSIV seat grinding system and a reactor vessel stud tensioner. In assessing the collective occupational radiological dose, each potentially significant dose-causing activity was evaluated. The major activities are provided in Table 12.4-1. Examples of significant design improvements that affect dose assessment in different plant areas are discussed below.

12.4.1 Drywell Dose

For the ESBWR drywell, design simplicity is the key to reduced occupational doses. Reactor systems are simpler with more passive safety features. The recirculation piping and pumps have been eliminated and a steel cylindrical shield has been provided around the reactor vessel to reduce drywell radiation fields.

Significant dose-causing activities identified for the drywell primarily involve maintenance tasks. Because the drywell is inaccessible during full power operation except to perform testing and maintenance, surveillance activities are non-existent. In addition, testing activities are at a minimum, except during and after equipment maintenance.

Projected ESBWR annual radiation exposures and historical typical BWR annual radiation exposures are shown in Table 12.4-1.

The major drywell activities identified in Tables 12.4-1 are:

- Main Steam Isolation Valve (MSIV) Repair
- Safety Relief Valve (SRV) Work
- Fine Motion Control Rod Drive (FMCRD)/Automated Fixed In-Core Probe (AFIP) Work
- Local Power Range Monitor (LPRM) Work
- In-Service Inspection (ISI)
- Misc. Valves
- Misc. Instrumentation

The Nuclear Boiler System (NBS) supplies steam to the main turbine. The MSIVs are located in the upper drywell area (4 valves) and in the reactor building outboard of the primary containment isolation wall (4 valves).

The ESBWR design incorporates three specific features to reduce occupational exposures in the MSIV maintenance areas:

- improved MSIV leakage rate test procedures,
- improved maintenance procedures with some procedures automated, and
- reduced radiation fields, primarily due to the absence of the recirculation piping.

The MSIVs require periodic testing and maintenance to ensure proper action and leak tightness. Maintenance operations incorporate an automatic seat grinding system and other special tools. Overall maintenance is reduced by use of the MSIV overhauling devices, use of main steamline plugs and the automatic MSIV grinding system. Use of these automatic systems results in an additional overall reduction in maintenance times of approximately 50%. This, along with improved drywell access, significantly reduces the maintenance time necessary for MSIV repair.

Beginning in the early 1980's, the BWR Owner's Group began an extensive study of the causes for failure of MSIVs to meet the leakage rate limits and the extensive person-hours required to maintain these valves. As a result of these studies, the ESBWR uses the improved leakage rate test procedures and latest technology for valve maintenance to reduce the personal exposures.

As a result of these aids, there is an estimated overall maintenance person-hour reduction to the value shown in Table 12.4-1.

Early studies on dose rates during MSIV maintenance showed increases in dose rate directly proportional to recirculation line activity. The ESBWR does not have these recirculation lines, thus removing the most significant shutdown source of radiation in the drywell. Additionally, the ESBWR is designed to limit the use of cobalt bearing materials on moving components that have historically been identified as major sources of in-water contamination. Overall, the feedwater line radiation is expected to be a factor of two lower than current BWRs.

The estimated dose rate for SRV work is shown in Table 12.4-1. In the ESBWR, the primary source of radiation exposure, the recirculation lines, has been removed. Overall, the reduction in drywell dose level for these types of maintenance is shown in Table 12.4-1. Overhead tracks and in-place removal equipment is provided in the ESBWR to reduce dose rates for maintenance.

A design improvement for the Neutron Monitoring System (NMS) involves replacing the conventional Traversing In-core Probe (TIP) system with fixed in-core detectors (AFIP) for calibrating the Local Power Range Monitors (LPRMs). Eliminating the complex drive and indexer mechanism with associated controls, which are required to insert and withdraw the TIPs from the core region, substantially improves operability, maintainability and reduces occupational radiological exposures.

LPRM design has been improved and currently each monitor lasts up to seven years. The estimated annual time and dose rates are shown in Table 12.4-1.

The drywell design includes many features to accommodate in-service inspection (ISI). Some of these include the use of stand-off mirror type insulation around the reactor vessel, use of remote-operated mechanical devices for inspection of the RPV body and nozzle welds, removable pipe insulation, and provisions for additional ISI operations laydown space. The natural circulation simplifies the design within the drywell by eliminating the recirculating loops, pumps, pipe supports, hangers, and shock suppressors. This also results in reduced ISI maintenance and personnel exposures.

ISI primarily consists of NDE examination of vessel and piping systems and welds. ESBWR ISI is estimated based on the following:

- Elimination of recirculation lines and pumps with savings of annual time and dose.
  - Elimination of 14 nozzle inspections at 2 per year.
  - Elimination of shield penetration and shield plug removal.
  - Reduction in drywell dose by 50% with the provision that the feedwater line dose is more than half the recirculation line dose and general drywell dose level and therefore, removal of recirculation line inspection may reduce the general drywell dose rate by 50%.
- Overall, it is estimated that by use of automated inspection, person-hour expended in ISI is reduced by a factor of two.
- The total vessel weld length inspection is reduced and the total weld inspection is reduced from the inspection required for conventional BWRs.

- The ESBWR design incorporates specific access into inspection areas past insulation areas with additional reductions in annual time.

The overall person-hours and typical effective dose rate for the ESBWR are shown in Table 12.4-1.

Simplified systems result in a significant reduction in the total number of valves and instrumentation located in the drywell with an accompanying decrease in maintenance time. Valve design is also enhanced. For example, operation of the Gravity-Driven Cooling System (GDCS) requires reactor depressurization. This depressurization utilizes eight depressurization valves (DPVs) with four located on the steam pipes and four located on stub tubes off the RPV shared with the IC lines in the upper drywell. Squib valves were selected for the DPV function because they are simple, reliable, eliminate all leakage concerns and have low maintenance requirements. The DPVs are a non-leak, non-simmer, and non-maintenance design. They also simplify the Automatic Depressurization System (ADS) by reducing the total number of relief valve discharge lines and steam quenchers mounted in the suppression pool. The estimated annual time to perform maintenance on miscellaneous drywell valves is shown in Table 12.4-1. The estimated annual time to perform maintenance on miscellaneous drywell instrumentation is also shown in Table 12.4-1.

Improved materials of construction and design for the reactor core fuel reduces the probability of fuel failure, thus, fuel leakage is significantly limited. This results in a reduced source term throughout the facility's radiation areas. Dose-reduction improvements also include improved water chemistry and the use of special materials for the reactor cooling water boundary. The design employs materials and processes that prevent intergranular stress corrosion and corrosion cracking by adopting resistant materials, limiting sensitizing operations, incorporating heat treatment techniques and eliminating crevice conditions. Also, there is reduced radiation fluence to materials and thereby, lower stresses experienced by materials typically exposed to high fluence. This fluence creates radioactive corrosion products from materials located in the reactor core and deposited on the fuel. Reduced radiation fluence decreases the activated corrosion products available for transport and plateout on out-of-core surfaces. In addition, a significant reduction in the drywell radiation levels result by restricting the cobalt content of selected vessel internals, using materials or cladding with corrosion resistance, and designing for erosive conditions.

Other drywell work includes items such as minor valve maintenance and instrumentation work. Overall reduction, in this effort, to the values shown in Table 12.4-1 is estimated due to the following ESBWR design improvements:

- Significant savings in total hours are estimated due to removal of the recirculation lines with miscellaneous recirculation line work such as line snubbers, fewer drywell cooling units, and less assembly/disassembly work on insulation.
- Overall reduction in the drywell radiation due to removal of the recirculation system results in the reduction of the overall upper drywell dose rate to the values shown in Table 12.4-1 because the components involved, such as drywell coolers, typically do not carry radioactive inventory.

12.4.2 Reactor Building Dose

The reactor building surrounds the RCCV and provides holdup and decay of radionuclides during an accident. It has been arranged to take advantage of the reduced quantity of equipment associated with the simpler reactor systems. The building arrangement features numerous dose-reducing benefits and improved equipment maintenance times. Equipment is more accessible which facilitates improved access control and maintenance. The building features enhanced accessibility on all floors. Equipment access is provided for all surveillance, maintenance and replacement activities with local service areas and laydown space for periodic inspections. Lifting points, monorails and other installed devices are provided to facilitate equipment handling and minimize the need for re-rigging individual equipment movements. Valve galleries are provided to minimize personnel exposure during system operation or preparation for maintenance.

Projected ESBWR annual radiation exposures are shown in Table 12.4-1.

The major reactor building activities identified in Tables 12.4-1 are:

- Reactor Pressure Vessel (RPV) Access/Reassembly
- Refueling
- Control Rod Drive Hydraulic Control Unit (CRD HCU) Work
- Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) System
- Instrumentation
- Other

Refueling operations involves all work with fuel and reactor components performed above the reactor and in pool area. Reactor vessel access and reassembly exposure times are reduced by use of a special stud tensioner for the RPV head bolts. The projected time to remove the RPV bolts with this equipment and reassemble, and an average estimated exposure are shown in Table 12.4-1. Underwater transfer for the dryer and separator decreases exposures during refueling operations.

Refueling exposures are also decreased by use of an automated refueling platform. The improved fuel, inspection equipment and increased remote operations significantly reduce the refueling floor exposure. Also, fuel sipping is not required based upon the improved fuel design fuel.

The RWCU/SDC purifies reactor coolant during normal operation and shutdown. Two 100% redundant trains are provided in the ESBWR design that uses state-of-the-art water treatment technology to significantly reduce the concentration of radionuclide material in the coolant. In addition, the material of construction for the system is stainless steel for those portions in contact with the reactor coolant. For system piping, smooth bends are used instead of welds and the nuclear grade pipes are electro-polished to reduce corrosion product buildup.

RWCU/SDC maintenance work consists of inspection for two pumps per year in each train. The ESBWR uses canned pumps in both trains with an estimated reduction in maintenance time to the values shown in Table 12.4-1. With improved water chemistry and overall reductions in reactor water concentrations due to the two percent cleanup system, the effective dose rate is estimated at the value shown in Table 12.4-1.

The ESBWR should reduce the typical person-hours per year with significant reductions in instrumentation due to reduced emphasis on active safety systems in lieu of passive systems, and combining systems such as the FAPCS, or deleting systems such as the TIP system. With improvements in water chemistry systems, the ESBWR should be able to reduce the effective dose rate to the value shown in Table 12.4-1.

Simplified systems in the reactor building result in a significant reduction in the total number of valves with an accompanying decrease in maintenance time. This work includes all valve work, minor maintenance, and CRD hydraulic line work. Use of live-load valve packing to control stem leakage reduces maintenance and worker radiation exposure for valve repair. The major task in this area is the hydraulic control units. With the use of the FMCRD units, an additional reduction of maintenance is anticipated. In addition, the ESBWR reactor building has been designed to provide for ease of maintenance with overhead lifts, coordinated hatchways and ample space to maintain in-place equipment. In addition, most of the equipment in the reactor building is removable. Those pieces assembled in bundles shall be moved from within the drywell. Because of these factors, an additional reduction in work is anticipated, resulting in the final value shown in Table 12.4-1. Because of the improved water chemistry, the overall effective dose rate is anticipated at the reduced value shown in Table 12.4-1.

12.4.3 Fuel Building Dose

The Fuel building houses the Spent Fuel Pool and the FAPCS System that purifies spent fuel pool water during normal operation and shutdown. Simplified systems result in a significant reduction in the total number of valves with an accompanying decrease in maintenance time.

ESBWR refueling is performed as described above and expected doses in the Fuel building during refueling are shown in Table 12.4-1

12.4.4 Turbine Building Dose

The steam and power conversion system includes the turbine main steam system, the main turbine generator, main condenser, main condenser air removal system, turbine gland seal system, turbine bypass system, extraction steam system, condensate purification system, and the condensate and feedwater pumping and heating system. The heat rejected to the main condenser is removed by a circulating water system and discharged to the normal power heat sink.

Steam, generated in the reactor, is supplied to the high-pressure turbine and the second stage reheater of the steam moisture separators/reheaters. Steam leaving the high-pressure turbine passes through a combined moisture separator/reheater prior to entering the low pressure turbines. The moisture separator drains, steam reheater drains, and the drains from the two high pressure feedwater heaters are drained to the direct contact feedwater heater which is combined with a feedwater storage tank. The reactor feedwater pumps take suction from the direct contact feedwater heater storage tank. The low pressure feedwater heater drains are cascaded to the condenser.

Steam exhausted from the low-pressure turbines is condensed and deaerated in the condenser. The condensate pumps take suction from the condenser hotwell and deliver the condensate through filters and demineralizers, gland steam condenser, steam jet air ejector condenser, off-gas recombiner condensers, and through the low-pressure feedwater heaters to the direct contact feedwater heater storage tank. The reactor feed pumps discharge through the high pressure feedwater heaters to the reactor.

Projected ESBWR annual radiation exposures are shown in Table 12.4-1.

The major turbine building activities identified in Table 12.4-1 are:

- Turbine Overhaul
- Valves/Pumps
- Condensate Treatment
- Other

With additional operational improvements in automating and a simpler overall system design, the expected overall turbine maintenance work is reduced to the value shown in Table 12.4-1. The value shown in Table 12.4-1 is assumed for turbine overhaul work.

The condensate system in the ESBWR uses hollow-fiber filled filters that require approximately half the maintenance of typical systems, resulting in an estimated annual maintenance time shown in Table 12.4-1. The material of construction for the condenser tubesheet is titanium which reduces leakage of corrosion products into the feedwater. Low pressure feedwater drains from the feedwater heaters are cascaded back to the condenser; thus, all corrosion products from these drains are filtered via condensate filter/demineralizers before returning to the RPV. The overall effective dose rate is estimated at the value shown in Table 12.4-1.

12.4.5 Radwaste Building Dose

Radwaste Building work consists of pump and valve maintenance, shipment handling, radwaste management and general clean up activity. Radwaste building doses result from routine surveillance, testing, and maintenance of the solid and liquid waste treatment equipment. The liquid treatment system collects liquid wastes from equipment drains, floor drains, filter backwashes and other sources within the facility. The solid treatment system processes resins, backwash slurries and sludge from the phase separator. It also processes dry active waste from the plant. Some examples of radwaste activities include movement of casks and liners, filter handling, resin moving and installation and removal of mobile radwaste processing skids. Both waste treatment systems are based on current mobile radwaste processing technology and avoid complex permanently installed components. All radwaste tankage and support systems are permanently installed. More of the radwaste operations involve remote handling than in a typical BWR. This, as well as improved maintenance procedures and a more flexible radwaste system and building design, leads to the estimated value shown in Table 12.4-1 for maintenance tasks in the Radwaste Building. The average dose rate shown in Table 12.4-1 is estimated for all operations.

12.4.6 Work at Power Doses

Routine work at power represents various tours of operators through the plant each shift, inspections and miscellaneous maintenance in radiation zones, as necessary. It covers all aspects of plant maintenance performed during normal operations from health physics coverage to surveillance, to minor equipment adjustment and repair. Overall, the ESBWR is designed using more automated and remote handling equipment. It is estimated there is a reduction in the total hours for work at power to the value shown in Table 12.4-1.

Exposure from these miscellaneous surveillance, testing and maintenance activities at power is due to N-16 as well as reactor coolant corrosion and fission products. Additional shielding is provided to reduce radiation levels in routinely occupied areas during power operation from N-16 sources. The ESBWR is expected to have lower general radiation levels as compared to the typical BWR due to more stringent water chemistry controls, a full-flow condensate flow system, a 2% reactor water clean up program, titanium condenser tubes, and low cobalt usage. The overall estimated effective dose for work at power is shown in Table 12.4-1.

12.4.7 COL Information

None.

12.4.8 References

None.

Table 12.4-1

Projected ESBWR Annual Radiation Exposure

Facility Area/Task	Estimated Annual Time, person-hour	Estimated Average Dose Rate, µSv/hr (mrem/hr)	Projected Annual Collective Dose, person-Sv (person-rem)
Drywell			
MSIV Repair	1,000	90 (9.0)	0.090 (9.0)
SRV Work	600	135 (13.5)	0.081 (8.1)
FMCRD/AFIP	350	170 (17.0)	0.063 (6.3)
LPRM Work	100	500 (50.0)	0.050 (5.0)
ISI	840	120 (12.0)	0.101 (10.1)
Misc. Valves	1,000	50 (5.0)	0.050 (5.0)
Misc. Instrumentation	500	50 (5.0)	0.025 (2.5)
Reactor Building			
RPV Access/Reassembly	1,200	30 (3.0)	0.036 (3.6)
Refueling	1,000	12 (1.2)	0.012 (1.2)
CRD HCU	120	45 (4.5)	0.006 (0.6)
RWCU/SDC	200	90 (9.0)	0.018 (1.8)
Instrumentation	600	50 (5.0)	0.030 (3.0)
Other	3,400	28 (2.8)	0.095 (9.5)
Fuel Building			
Refueling	1,000	5 (0.5)	0.005 (0.5)
FAPCS	200	90 (9.0)	0.018 (1.8)
Turbine Building			
Turbine Overhaul	12,000	3 (0.3)	0.036 (3.6)
Valves/Pumps	700	70 (7.0)	0.049 (4.9)
Condensate Treatment	1,000	70 (7.0)	0.070 (7.0)
Other	6,000	1 (0.1)	0.006 (0.6)
Radwaste Building	1,000	45 (4.5)	0.045 (4.5)
Miscellaneous Work at Power	550	80 (8.0)	0.044 (4.4)
Total	32,790		0.930 (93 person-rem)

12.5 OPERATIONAL RADIATION PROTECTION PROGRAM

12.5.1 Objectives

The ESBWR design includes health physics facilities and features providing capabilities for administrative control of:

- the activities of plant personnel to limit personnel exposure to radiation and radioactive materials as low as reasonably achievable (ALARA) and within the guidelines of 10 CFR 20.
- effluent releases from the plant to maintain the releases ALARA and within the limits of 10 CFR 20 and the plant Technical Specifications.
- waste shipments from the plant to meet applicable requirements for the shipment and receipt of the material at the storage or burial site.

12.5.2 Equipment, Instrumentation, and Facilities

The health physics facilities are located in the Service building. Access to the radiologically controlled areas of the Reactor, Fuel, Turbine, and Radwaste buildings is normally through the entry/exit area of the health physics facilities of the Service building. Exit from the radiologically controlled areas is at the same location.

The health physics area contains the personnel contamination monitoring equipment, decontamination shower facilities, changing rooms and first-aid equipment. The changing rooms are provided with lockers, wash sinks, showers and toilet facilities.

Portable radiation survey instrumentation is stored at the access control and health physics room and at in-plant control points. This instrumentation allows plant personnel to perform radiation, contamination and neutron surveys, as needed, as well as to collect samples for airborne analysis. Shielded rooms are provided in the health physics area for radioactivity analysis and for calibration of survey instruments.

The portable instrumentation is out of the ESBWR standard plant scope. See Subsection 12.5.4 for COL license information. The non-portable airborne radiation monitoring equipment is described in Subsection 12.3.4.

12.5.3 Operational Considerations

Out of the ESBWR Standard Plant Scope. See Subsection 12.5.4 for COL license information

12.5.4 COL License Information

12.5.4.1 Radiation Protection Program

COL applicants will provide, to the level of detail required by Regulatory Guide 1.70, the implementation of a radiation protection program for operational considerations.

12.5.4.2 Equipment, Instrumentation, and Facilities

COL applicants will provide, to the level of detail required by Regulatory Guide 1.70, the health physics facilities description as well as the related equipment and instrumentation.

12.5.4.3 Compliance with Paragraph 50.34 (f) (xxvii) of 10 CFR 50 and NUREG-0737 Item III.D.3.3

COL applicants will provide the portable instruments in operating reactors that accurately measure radio-iodine concentrations in plant areas under accident conditions and will provide training and procedures on the use of these instruments in compliance with Paragraph 50.34 (f) (xxvii) of 10 CFR 50 and NUREG-0737 Item III.D.3.3.

12.5.5 References

None

12A. CALCULATION OF AIRBORNE RADIONUCLIDES

This appendix presents a simplified methodology to calculate the airborne concentrations of radionuclides in a compartment. This methodology is conservative in nature and assumes that diffusion and mixing in a compartment is basically instantaneous with respect to those mitigating mechanisms such as radioactive decay and other removal mechanisms. The following calculations need to be performed on an isotope-by-isotope basis to verify airborne concentrations are within the limits of 10 CFR 20.

- For the compartment, all sources of airborne radionuclides need to be identified such as:
  - Flow of contaminated air from other areas
  - Gaseous releases from equipment in the compartment
  - Evolution of airborne sources from water leaking from equipment or sumps.
- Second, the primary sinks of airborne radionuclides need to be identified. This will primarily be outflow from the compartment but may also take the form of condensation onto room coolers.
- Given the above information, the following equation will calculate a conservative concentration:

$$C_i = \frac{1}{V}\sum_j \frac{S_{ij}}{\left(\lambda_i + \sum_k R_{ijk}\right)}$$

where:

C_i = concentration of the ith radionuclide in the room

V = volume of room

S_{ij} = the jth source (rate) of the ith radionuclide to the room (these sources are discussed below)

R_{ijk} = the kth removal constant for the jth source and the ith radionuclide as discussed below.

λ_i = radionuclide decay constant

12A.1 EVALUATION PARAMETERS

The following parameters require evaluation on a case-by-case basis dictated by the physical parameters and processes germane to the modeling process.

S_{ij} is defined as the source rate for radionuclide i into the compartment. Typically, these source rates take the form of:

- Inflow of contaminated air from an upstream compartment. Given the concentration of radionuclide i, C_i, in this air and a flow rate of "r", the source rate then becomes $S_{ij} = r\ C_i$.

- Production of airborne radionuclides from equipment. This typically takes two forms: gaseous leakage and liquid leakage.

For gaseous leakage sources, the source rate is equal to the concentration of radionuclide i, C_i, and the leakage rate, 'r', or $S_{ij} = r\,C_i$.

For liquid sources, the source rate is similar but more complex. Given a liquid concentration ci and a leakage rate, 'r', the total release from the leak is $r\,C_i$. The fraction of this release that then becomes airborne is typically evaluated by a partition factor, P_f that may be conservatively estimated from:

Noble Gases

$P_f = 1$

All others

$$P_f = \frac{h_t - h_f}{h_s - h_f}$$

where:

h_t = saturated liquid enthalpy

h_f = saturated liquid enthalpy at one atmosphere = 180.17 Btu/lb

h_s = saturated vapor enthalpy at one atmosphere = 1150.5 Btu/lb

Therefore, the liquid release rate becomes, $r\,C_i\,P_f$.

R_{ijk} is defined as the removal rate constant and typically consists of:

- Exhaust rate from the compartment. This term considers not only the exhaust of any initially contaminated air but also any clean air that may be used to dilute the compartment air.
- In compartment filter systems. Such filter systems are treated by the equation:

  $R_{ijk} = (1-F_i) \cdot r_i$

where:

r_i = filter system flow rate

F_i = filter efficiency for radionuclide i

- Other removal factors on a case-by-case basis that may be deemed reasonable and conservative.

12A.2 EXAMPLE CALCULATION

Values used below are examples only and should not be used in any actual evaluation. This example will look at I^{131} in a compartment of 283 $m^3 = V$. First, all primary sources of radionuclides need to be identified and categorized.

Flow into the compartment equals 424 m^3 per hour with the input I^{131} concentration equal to 7.4×10^{-6} Bq/ml (from upstream compartments) or 0.90 Bq/sec. No other sources of air either contaminated or clean air are assumed.

The compartment contains a pump carrying reactor coolant with a maximum specified leakage rate of 5.7×10^{-7} m^3 per minute at 288°C.

- Conservatively, it can be estimated based upon properties from steam tables that under these conditions 44% of the liquid will flash to steam and become airborne. The assumption of 44% flashing at 288°C is extremely conservative. See Reference 12A-1 for a discussion of fission product transport. Along with the flashing liquid, it is assumed that a proportional amount of I^{131} will become airborne; therefore, $P_f = 0.44$.
- Assuming iodine concentrations for reactor water of 5.3×10^{-4} MBq/gm of I^{131}, it is calculated that the pump is providing a source of I^{131} to the air of 1.6×10^{-6} MBq/sec to the air. Water density assumed at 0.743 gm/cm^3 based upon standard tables for water at 288°C.

Second, the sinks for airborne material need to be identified, which in this example include only exhaust that is categorized as flow out of the compartment at 150% per hour or 4.2×10^{-4} per second.

Therefore, for an equilibrium situation, the I^{131} airborne concentration from this liquid source would be calculated from the equation

$$A=1/V(S_1 / (\lambda + R_1) + S_2 / (\lambda + R_2))$$

where:

V	=	volume of compartment = 283 m^3
S_1	=	source rate in Bq per second = 1.6 Bq/sec from liquid
S_2	=	source rate from inflow = 0.9 Bq/sec
λ	=	isotopic decay constant in units of per second = 9.977×10^{-7}/sec
$R_1 = R_2$	=	removal rate constant per second (exfiltration) = 4.2×10^{-4} per second.
A =		2.117×10^{-11} MBq/ml of I^{131}.

12A.3 COL INFORMATION

None.

12A.4 REFERENCES

12A-1 Paquette, et al, "Volatility of Fission Products During Reactor Accidents," Journal of Nuclear Materials, Vol. 130 Pg. 129-138, 1985.